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

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Report on Scenario Development
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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 3.

All PAMINA reports can be downloaded from http://www.ip-pamina.eu.

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Report on Scenario Development

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Executive Summary

The main objective of RTDC-3 is to develop methodologies and tools for integrated performance assessment (PA) for various geological disposal concepts. RTDC-3 acts as a link between the review work undertaken in RTDC-1 and the practical cases investigated in RTDC-4. The comprehensive overview of PA methodologies and experiences that is carried out in RTDC-1 will result in the identification of deficiencies in methods and tools for PA. In RTDC-3 development work is carried out for some of the topics identified in RTDC-1. The RTDC-3 component is divided in several work packages (WP). This report focuses on WP 3.1 “Scenario Development”.

WP 3.1 consists mainly of two complementary topics: (1) identification of scenarios based on safety functions and (2) development of stylized scenarios. Therefore, this deliverable report covers both safety functions and stylized scenarios.

In the framework of WP 3.1 a number of internal documents (milestones reports) were generated, which address explicitly the mentioned topics. It should be also mentioned, that several further internal documents were provided that describe e.g. the normal evolution of the disposal system and the compilation of features, events and processes (FEP) that might affect the evolution. This work forms the basis in different ways for the both topics e.g. the application of developed methodologies and for the quantification of identified scenarios.

This deliverable report compiles the essential milestone reports of WP 3.1 that represents the relevant output for scenario identification on the basis of safety functions and stylised scenarios. Therefore, the above-mentioned additional internal documents, albeit very important for the topics, will be not further addressed in this report. The structure of this document consists of two parts that presents the individual topics. Each part is divided further into subparts which provide the content of selected internal documents.

Part 1: Safety functions

Safety functions have the potential to overcome certain drawbacks of the multi-barrier approach. In the multi-barrier system each barrier can be associated with one or more safety functions. This association can be useful as a basis for the identification of scenarios. The safety functions provided by individual barriers explain the functioning of the disposal system in the case of the normal evolution scenario. One approach is to assume, on the basis of a FEP analysis for example, that the behaviour of the barriers can be altered, leading to various alternative evolution scenarios. In the altered evolution scenarios, safety functions provided by other barriers can become more important compared to the normal evolution scenario. In this way, safety functions will allow a more systematic approach to the identification of scenarios.

The first contribution (part 1.1) describes a method for systematic scenario identification and illustrates its application to the case of geological disposal of radioactive waste in the Boom Clay formation in NE Belgium. The proposed method starts from a list of uncertainties in the safety statements, which underpin the safety functions. It is examined which uncertainty can affect a safety function. Four altered evolution scenarios have been identified.

Another contribution (part 1.2) characterises an engineering approach used for managing complex projects. This approach was applied to derive safety functions and scenarios for the evaluation of safety of the disposal of spent fuel assemblies in carbon steel canisters surrounded by a bentonite buffer in a granite host rock. The main principle of the approach is to define safety functions on a top-down basis regarding the disposal system, subsystems and its components. Further steps refer to the hierarchical arrangement of safety functions and the formulation of requirements. Finally, the safety functions are analysed by considering all possible interactions with surrounding systems, subsystems and components on a bottom-up basis.

Part 2: Development of stylised scenarios

Stylised scenarios are commonly used when the evolution of the disposal system can be influenced by phenomena involving large uncertainties that cannot be quantified without undue speculation. A notable example of such a phenomenon is future human intrusion into the disposal system: owing to the large uncertainties, associated with modelling intrusion scenarios, rules are needed to guide the development of stylised scenarios.

The term “stylised” should not be misunderstood as “generic”: even with stylised scenarios, there is a need to consider site-specific conditions and design concepts. Stylised scenarios are complementary to the normal and altered evolution scenarios that can be developed using safety functions.

In terms of rules and guidelines for the development of stylised scenarios, the first work in this topic (part 2.1) presents a regulatory perspective on stylised human intrusion scenarios. The ICRP principles of protection and guidance on human intrusion are summarised. Various regulatory approaches to the treatment of human intrusion are discussed. Further, a detailed example is given of the incorporation of the ICRP recommendations on human intrusion into regulatory guidance in the UK.

The development of stylised human intrusion scenarios for different disposal systems and host rocks (plastic clay and salt) are described in the next work (part 2.2). An overview is given of existing regulations, guidelines and recommendations on the treatment of human intrusion in safety evaluations - this links closely with the work presented in part 2.1. Thereafter, several examples of the treatment of human intrusion scenarios in safety cases are presented.

For the case of disposal in a clay formation, a methodology has been developed that allows the identification of a set of stylized human intrusion scenarios. The methodology is based on a systematic analysis consisting of 3 steps: identification of the relevant human actions taking into account the considered host formation and disposal site, considerations about the intrusion time and the identification of possible exposure modes. Finally, the proposed methodology has been applied.

For the case of disposal in a salt formation the regulatory framework and several recommendations used as the basis for the development of stylised human intrusion scenarios are presented. Relevant initial situations or actions as a potential basis for stylised human intrusion scenarios are identified and discussed by considering current techniques and procedures. As a result, various indicators are identified that might serve as a basis for the evaluation of the detection probability of anomalies associated with the disposal system and the emplaced radioactive waste. Several cases of exploratory drillings penetrating different locations of the disposal system are analysed regarding their detection probabilities.

The consideration of stylised scenarios in a somewhat different way is the subject of the last presented work (part 2.3). In this work stylised scenarios are understood as scenarios initiated by events with very low probabilities. Only those events have to be taken into account, for which it is not possible to derive probabilities due to their very infrequent occurrence. Therefore, stylised scenarios describe the release of radionuclides into the environment from emplaced spent fuel in a granite host rock, as a consequence of events with very low probabilities. This contribution has a strong relation to part 1.2.

Content

Part 1: Safety Functions

Part 1.1: Identification of Altered Evolution Scenarios on the Basis of Safety Functions

(Prepared by J. Marivoet, SCK•CEN, Belgium)

Foreword

The PAMINA project (2006-2009) largely coincides with the development of the new safety assessment methodology by the Belgian radioactive waste management agency ONDRAF/NIRAS. This new safety assessment methodology will be applied for the first Belgian Safety Case, called Safety and Feasibility Case 1 (SFC 1), which is scheduled to be finalised in 2013.

At the start of the PAMINA project in 2006, it was already decided that safety functions will play an important role in the SFC 1. They will be applied for defining the safety strategy, repository development, identifying altered evolution scenarios, identifying performance indicators, structuring the safety case and communication of the functioning of a disposal system to various audiences.

During the development of the safety strategy, which will be applied for the SFC 1, in 2006-2007, it was considered necessary to introduce safety statement trees that underpin the safety functions. These safety statements will be used to structure the assessment basis of the SFC 1. In 2007, the development of a safety assessment methodology was started that makes use of the safety functions as well as of the much more detailed safety statements. The treatment of uncertainty in the SFC 1 will also be integrated within the safety assessment methodology. The elaboration of the methodology for the treatment of uncertainty is planned for autumn 2008 and the testing of its application for scenario identification in 2009. The finalisation of the safety assessment methodology for the SFC 1 is scheduled by the end of 2009.

As the time schedule for the development of SFC 1 safety assessment methodology does not match with the planning of PAMINA, the present report is based on a number of preliminary elements of that methodology.

1 Introduction

Scenario development is defined as "the identification, broad description, and selection of potential futures relevant to safety assessments of radioactive waste repositories" (NEA, 2001). Its main objective is to show in a traceable and transparent way that all potentially important features, events and processes of the repository system as well as the main uncertainties have been taken into account in the safety assessment.

Two important aspects can be distinguished in scenario development:

- identification of the main evolution scenarios aiming at identifying a sufficiently representative set of scenarios: in most safety assessments a reference (also called normal evolution, expected evolution, or main) scenario is supplemented by a set of altered evolution scenarios;

- description of the identified scenarios in such a way that it is clearly shown how the main features, events and processes are treated within the considered scenario.

In recent years many efforts have been devoted to demonstrating that all relevant FEPs (features, events and processes) have been accurately treated within the safety case. This has led to the development of well documented methods for scenario descriptions. A good example is the approach used by SKB in its SR 97 safety analysis report (SKB, 1999), which is based on a detailed system description (internal processes, interactions illustrated in THMC diagrams) and a description of the initial state of the repository.

The identification and selection of the altered evolution scenarios is often far less systematic or documented than the scenario descriptions. In the US, logic diagrams were used for the construction of scenarios for the WIPP and Yucca Mountain sites (Guzowski, 1991; Swift et al., 2001). In the nineties, methods for systematic scenario identification based on repository "states" have been developed. Examples are the SKI/SKB approach (Andersson et al., 1989) and the "PROSA" methodology (Prij, 1993; Gomit et al., 1997). In these methods the repository system is divided into a number (n) of main components. Each main component can be in two (well functioning or not-well functioning) or three (good functioning, partially functioning or not functioning) states. Thus, the repository system can be in n^2 or n^3 states. Scenario initiating FEPs are then classified into one of the states of the repository system; each state corresponds to a family of possible altered evolution scenarios.

During the second half of the nineties, the application of the defence-in-depth concept in long-term safety assessments of radioactive waste disposal led to the introduction of safety functions. It was considered necessary to complement the multi-barriers principle with a set of safety functions that are provided by diverse mechanisms and components of the repository system.

In this report we propose a method for scenario identification in which the states of the repository system are based on the state (available or not available) of the safety functions (instead of the components in the SKI/SKB and PROSA approaches).

Recently (i.e. October 2006) SKB has published the synthesis report of the SR-Can project (SKB, 2006). In this report a method based on the potential loss of safety functions is applied for the selection of additional scenarios. The primary safety function in the KBS-3 repository concept of SKB is what they call *isolation* (this safety function is called *containment* in various other safety cases).

Another safety function *limit advective transport* is provided by the buffer. The selection of additional scenarios in the SR-Can project is based on combining failed states of the two before mentioned safety functions.

The methodology that is proposed in Chapter 2 can, in principle, be applied to all types of repositories. The proposed methodology is applied in the present report to a repository for the disposal of high-level and long-lived radioactive waste that is assumed to be excavated in the Boom Clay formation in NE Belgium. The considered repository system is briefly described in Chapter 3. Chapter 4 explains the derivation of the safety functions. Chapter 5 gives a description of the functioning of the repository system in the case of the expected evolution scenario. Safety statements are introduced in Chapter 6. The treatment of uncertainty is discussed in Chapter 7 and its application for scenario identification in Chapter 8. Conclusions are drawn in Chapter 9.

2 Methodology

The proposed methodology for scenario identification consists of 7 consecutive steps.

Step 1: Define a set of safety functions for the considered disposal system. On the basis of the experience gained from previous safety assessments it is possible to identify a set of safety functions that explain how diverse physical and chemical processes provided by the barriers of the repository system ensure that the radionuclides present in the disposed waste will be contained in the repository system during the desired time spans.

Step 2: Develop a safety concept based on the functioning of the disposal system in the case of the reference scenario. This is strongly directed by the question "when is a safety function expected to be available or when can it be relied upon". All the safety functions provided by the repository system do not have to function at the same time. The availability of one safety function, e.g. containment (isolation in SKB's terminology) provided by a watertight overpack, can make that the other safety functions are not needed during that period.

Step 3: Build a structured set (tree) of safety statements. These statements are derived from the requirements on the disposal system, on the sub-systems and on individual components. The top level safety statements correspond to the safety functions, possibly extended with statements concerning the initial quality of the engineered barriers and the dilution (which is mainly provided by the environment of the repository system, e.g. the aquifers surrounding the host formation) and the biosphere.

Step 4: Make a systematic analysis of the uncertainty affecting the safety statements.

Step 5: Identify a first list of possible altered evolution scenarios by considering all identified uncertainties and by testing if they have the potential to propagate to higher level statements, and eventually to affect the safety functions.

Step 6: Derive a final set of altered evolution scenarios. This is done by constructing functional diagrams illustrating the impact of the considered uncertainty in a safety statement on the functioning of the disposal system and by grouping, as far as possible, scenarios with identical or strongly similar functional diagrams.

3 Considered repository system

The considered repository system is assumed to be constructed in the Boom Clay formation in NE Belgium. At the well-studied site of Mol, the host formation is about 100 m thick, of which an 80 m thick central zone consists of very homogeneous clay. A scheme of the geology at the Mol site is given in Fig. 1.

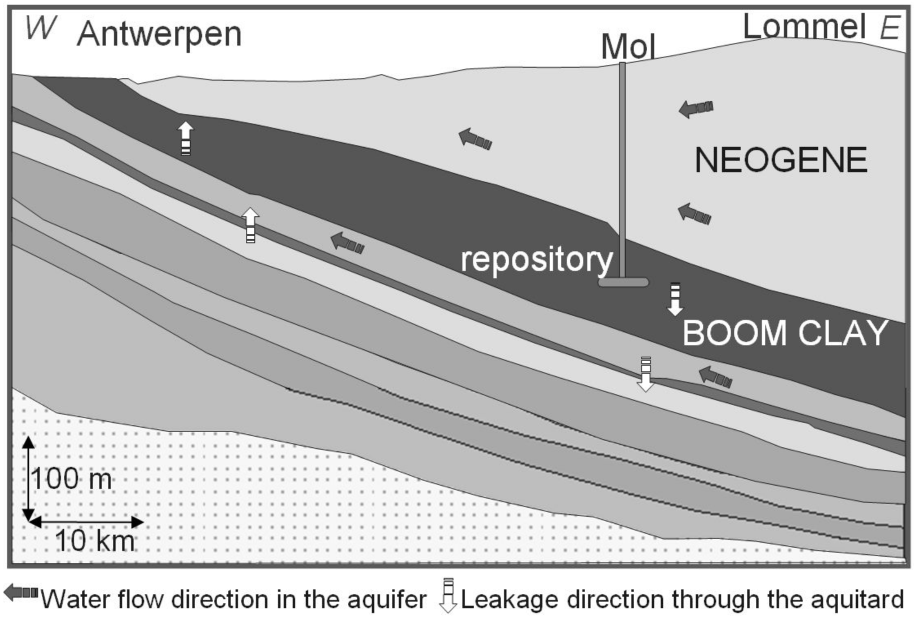


Fig. 1: Scheme of the geological structure at the Mol site; the 180 m thick sandy formation of the Neogene overlies the considered host clay formation.

The considered repository will have a central access facility consisting of at least two vertical shafts and two transport galleries. The disposal galleries will be excavated perpendicular upon the transport galleries (Wickham, 2008). Because the Boom Clay is a plastic clay, a concrete gallery lining is required to avoid convergence of the gallery walls. A sketch of the different components in the envisaged repository system is shown in Figure 2.

The high-level waste (HLW) canisters and spent fuel overpacks will be packed in so-called "supercontainers". This design is based on the contained environment concept, which is aimed at establishing and maintaining a chemical environment around the overpack that should guarantee the fulfilment of the "engineered containment" safety function during the envisaged timeframe. In one supercontainer, two HLW canisters, four uranium oxide (UOX) spent fuel assemblies, or only one assembly in case of mixed oxide (MOX) fuel (in order to obey thermal restrictions at the overpack level), are placed within a carbon steel overpack. The spent fuel assemblies are first inserted into carbon steel boxes for ease of handling, and these may be filled with sand to limit the amount of water around the assemblies, what will avoid the occurrence of criticality, and/or with an inert gas to protect against corrosion. The boxes will be supported within the overpack by a carbon steel or cast iron basket. The 30 mm thick carbon steel overpack will then be placed into a prefabricated cementitious cylindrical buffer (called "phase 1 concrete"), possibly enclosed by a 6 mm thick stainless steel envelope. The buffer concrete will be based on Ordinary Portland Cement (OPC), which will provide the highly alkaline environment to keep the overpack passivated. The cylindrical cavity between the overpack and the buffer will be filled with a concrete filler ("phase 2 concrete"). The top of the buffer is closed by pouring concrete which forms the sealing plug ("phase 3 concrete"). The resulting integrated waste package, manufactured in surface buildings, is called a "supercontainer". The supercontainers will be placed end-to-end in the centre of the disposal galleries. A cementitious backfill will be placed between the supercontainers and the gallery walls. As an example, Fig. 3 shows a scheme of a disposal gallery configuration for uranium oxide spent fuel. Figure 4 shows the cross-sections of the supercontainer concepts for vitrified HLW and UOX and MOX spent fuels.

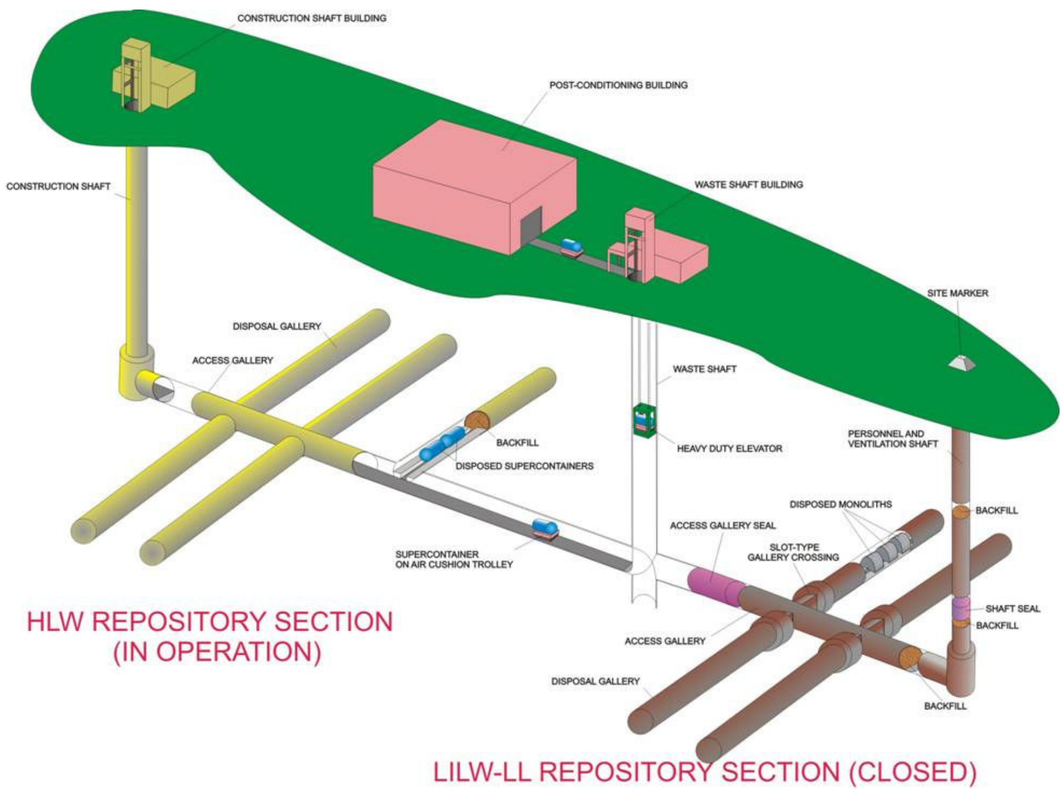
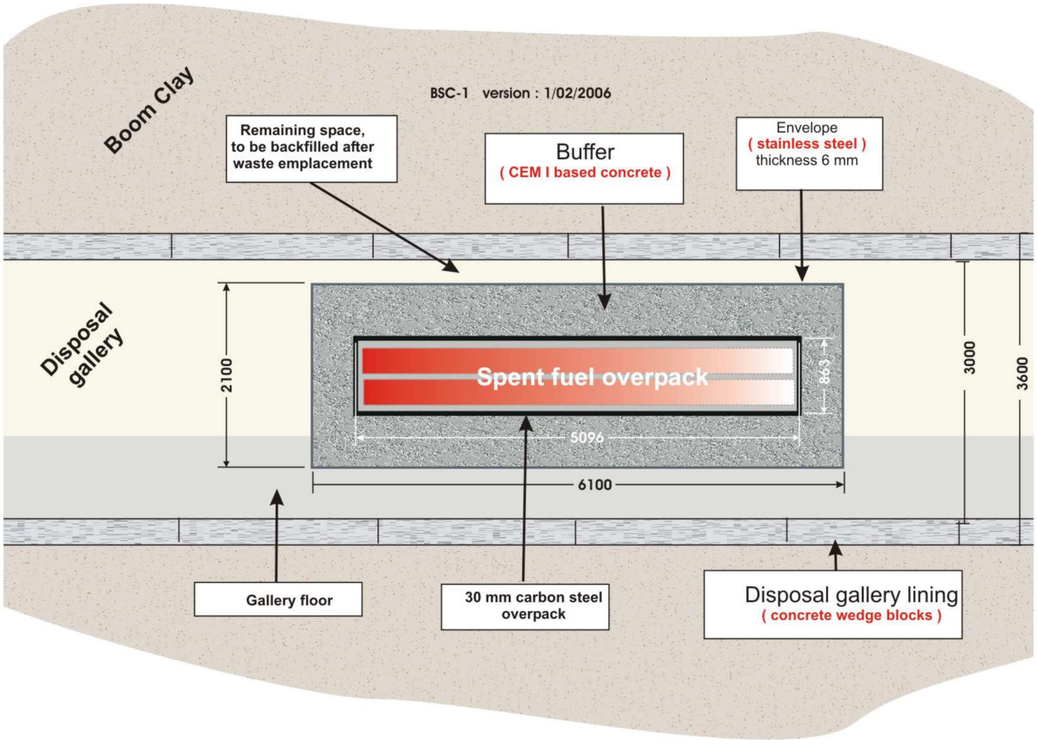
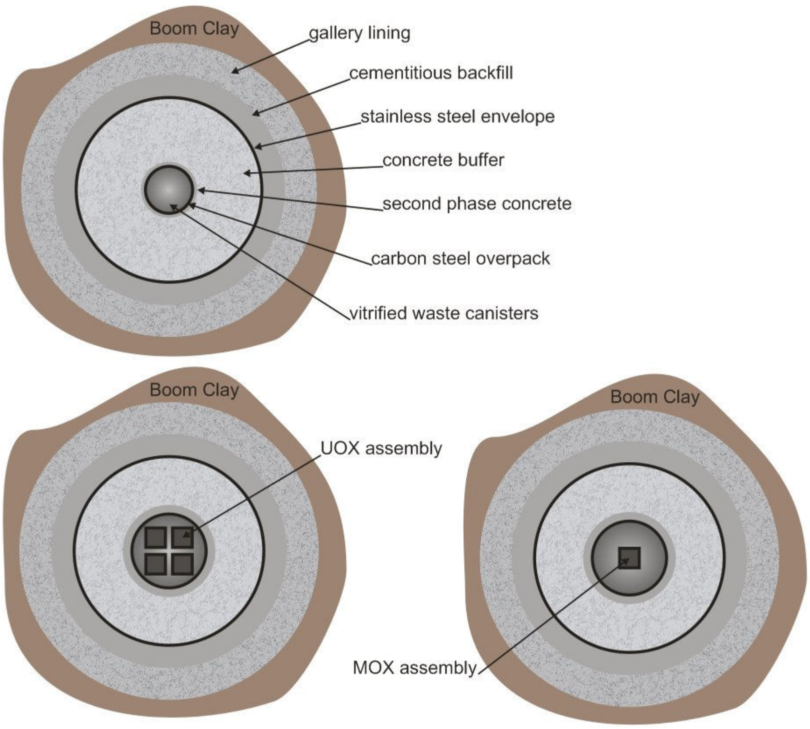


Fig. 2: Lay-out of a geological disposal system for high-level (vitrified HLW and spent fuel) and long-lived waste.



1.1.1.1 Fig. 3: Schematic diagram showing a longitudinal section through a disposal gallery containing a supercontainer with UOX spent fuel assemblies.



1.1.1.2 Fig. 4: Cross-section of the supercontainer concept for high-level waste streams existing at present in Belgium.

4 Derivation of safety functions

4.1 Introduction

The first step of the proposed methodology consists in defining a set of safety functions for the considered disposal system. The starting point for the development of safety functions related to a geological repository is 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). This means that the radionuclides present in the disposed waste are, as long as possible, kept concentrated in a waste package and confined in the repository system.

The derivation of safety functions is based on two considerations:

a) the barriers provided by the disposal system have to protect man and the environment from possible harmful effects due to the radionuclides present in the disposed waste;
b) to ensure that the containment will be maintained during a very long period, the system of engineered and natural barriers provided by the repository system has to be isolated from external processes and events that might weaken the performance of the barrier system.

The management strategy to "concentrate and contain" radioactive waste constitutes a major starting point for the development of a disposal system. To ensure the required protection of man and the environment, its objectives are to prevent that:

c) man can come into direct contact with toxic substances present in the waste, i.e. preventing exposure as a result of man reaching the waste (intrusion) or of insufficient shielding of the waste from man;
d) the contaminants in the waste are released from the disposal system and dispersed into the environment, i.e. avoiding exposure due to toxic substances that reach the human environment.

A repository system based on multiple safety functions will be required to ensure post-closure safety and the disposal system developer must make sure that post-closure safety is not unduly dependent on a single component or function.

Hereafter we give a description of the set of safety functions that has been developed within the Belgian high-level waste disposal programme of ONDRAF/NIRAS (2007).

To achieve the first objective, the waste must be isolated durably from the human environment (biosphere) by placing it into a difficult-to-access, stable and well-shielded place. This is realised by the long-term safety function "isolation", which is described in Section 4.2.1.

The second objective can be achieved by trying to prevent any dispersion of the contaminants from the disposed waste. This is realised by the long-term safety function "engineered containment", which is presented in Section 4.2.2.

As prevention of dispersion of contaminants from the waste cannot be fully guaranteed over the total period that the waste presents a hazard, the logical subsequent step is to limit and retard dispersion of contaminants towards the environment as much as achievable or needed, so that the release of contaminants from the disposal system into the environment remains at all times limited to an acceptable level. This is realised by the long-term safety function "delay and attenuate the releases", which is described in Section 4.2.3.

4.2 Definitions of the safety functions

4.2.1 Isolation (function I)

The first safety function of the disposal system is to isolate the waste (i.e. radionuclides and chemical contaminants) durably from man and his environment to prevent direct access to the waste and to protect the disposal facility from the effects of surface processes. This isolation function can be achieved by placing the waste at a stable location, where inadvertent human intrusion is unlikely.

The dual objective of the "isolation" function leads to the definition of two sub-functions:
a) Reduce the likelihood and possible consequences of inadvertent human intrusion (sub-function I1)

This sub-function aims at limiting the likelihood of inadvertent human intrusion into the disposal facility and, in case such an intrusion would occur, at limiting its possible consequences. Siting the repository away from areas of underground resources contributes to reducing the likelihood of inadvertent intrusions in the disposed waste.
b) Create stable conditions for the disposed waste and the system components (sub-function I2)

The aim of this sub-function is to place the waste in a location that is shielded from changes and perturbations occurring at or near the surface, such as, e.g., climate changes or relatively rapid changes in chemical and physical conditions at the surface. As such, this sub-function creates a stable physico-chemical environment for the components of the disposal system that contribute to the fulfilment of the other two safety functions (function C and R).

A durable isolation requires, among others, a stable geological setting to avoid or to limit geomorphological processes leading to denudation of the waste, for instance in the case of deep disposal owing to "*uplifting"* of the geological formations and/or to erosion phenomena.

4.2.2 Engineered containment (function C)

The second long-term safety function of a disposal system "engineered containment" attempts to contain as long as possible all the contaminants inside the waste package. The aim of this safety function is to prevent contact between the contaminants present in the waste form and the infiltrating or already present water (which is the main vector of dispersion of the contaminants) by the presence of impermeable barriers around the waste form. As long as this function is operational, dispersion of contaminants from the disposed waste does not occur and benefit is taken of the radioactive decay of the radionuclides within the waste form to reduce the total radiological impact.

However, the containment of the contaminants within the waste form and its watertight overpack cannot be fully guaranteed for the whole period during which the waste constitutes a hazard. Therefore, an additional safety function is necessary.

4.2.3 Delay and attenuate the releases (function R)

In case of partial or complete failure of the "engineered containment" function (which is inevitable in time), there is another function that retards and limits the release of contaminants from the disposal system. This is the function of "delay and attenuate the releases", which aims at retaining as long as possible the contaminants within the disposal system, by reducing the quantity of contaminants migrating and ultimately leaving the disposal system and by limiting the migration and release rates. As long as the contaminants are not released from the disposal system into the environment, containment on the repository system level is effective.

Delaying the releases takes benefit from the radioactive decay of the radionuclides during their long residence times, which result from the very slow migration through the main engineered and natural barriers. The spreading in time of the contaminant fluxes released from the disposal system makes that the release rates of the residual (i.e. non-decayed) activity or the contaminants from the disposal system into the environment are extremely low.

Here, three sub-functions, which contribute to the delay and the attenuation of the releases, can be defined:

a) Limit the release from the waste form (R1)

With this sub-function, the system limits the release of contaminants from the waste form, and the radionuclide release out of the waste container and overpack, in which they are enclosed, thus spreading this release in time. Various physico-chemical processes contribute to the resistance to leaching of the contaminants from the waste form, such as slow dissolution mechanisms and low solubility. Other phenomena can also limit the quantity of contaminants released per time unit: the spreading in time of the failure of the waste containers or overpacks (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).

b) Limit the water flow through the system (R2)

With this sub-function, the flow of water through the system is opposed, resulting in limited quantities of water infiltrating or moving through the system due to the presence of low permeability barriers. The limited quantity of infiltrating or moving water will prevent or limit advective transport of contaminants after their release from the waste form, resulting in a contaminant transport through the disposal system that is mainly controlled by diffusion. The slow diffusive transport of the contaminants through the engineered and natural barriers of the disposal system will considerably spread in time the release of the contaminants from the disposal system. This function determines also the amounts of water that actually come into contact with the barriers that fulfil the engineered containment function (C) or the slow release from the waste form sub-function (R1).

c) Retard contaminant migration (R3)

The system fulfils this sub-function by the slow diffusive transport already mentioned under sub-function R2. Furthermore, processes such as sorption, isotopic exchange, and contaminant (co)precipitation will also contribute to retarding and spreading in time the radionuclide releases from the system.

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

The environment of the disposal system must also be taken into account when assessing the safety and impact of a disposal system, especially when calculating the doses or risks for man, or the concentration of contaminants in different parts of the environment. The contribution of the environment to long-term safety mainly consists in diluting and dispersing the contaminants released from the disposal system, because the impact on man and the environment is directly proportional to the contaminant concentrations. This is the dispersion and dilution role (D) of the environment. Compared with the safety functions of the disposal system itself, this role can only be of secondary importance, for the following reasons. First, all efforts made to maximize or optimize this role would lead to a "disperse and dilute" strategy, instead of the chosen "concentrate and confine" strategy. Secondly, it is more difficult in the long term to rely heavily on this contribution of the environment, considering that it can change considerably with time, especially with the changing climate. For these reasons, safety functions are only defined for the disposal (barrier) system itself and not for its environment.

5 Functioning of the barrier system in case of the reference scenario

The safety functions defined in Section 4.2 can now be applied to explain the functioning of the disposal system in case of the reference scenario. This is illustrated in Fig. 5 for the disposal of vitrified high-level waste.

During the thermal transient phase of the repository the engineered containment function (C) is effective; this function is fulfilled by the overpack. A few thousands of years after the end of the thermal phase, the overpacks are expected to lose their integrity because of corrosion. From then, the safety function delay and attenuate releases (R) becomes effective (before overpack corrosion, this safety function was present but not effective, because no groundwater was in contact with the waste matrix). The sub-function limited release (R1) is partially fulfilled by the glass matrix, which is expected to have a lifetime of a few tens of thousands of years. Solubility limits also contribute to the limitation of the release of many radionuclides from the waste package. This process can be much longer active than the matrix lifetime; the period during which solubility contributes to safety function R1 depends on the inventory of the considered element in the waste and the solubility limit. The sub-functions limit water flow (R2) and retard migration (R3) also become effective after overpack perforation. These 2 sub-functions are fulfilled by the buffer and the clay host formation; they are expected to remain effective during the (geological) system containment phase. The safety function isolation (I) is expected to be effective during the whole lifetime of the repository system.

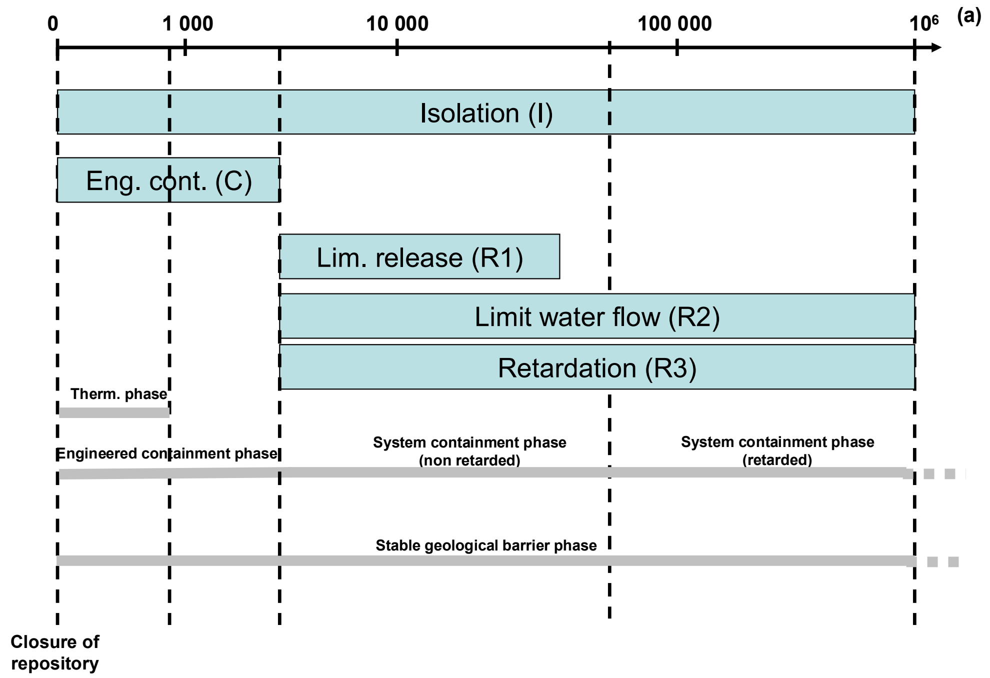


Fig. 5: Functional diagram of the repository system for the disposal of vitrified high-level waste in the case of the reference scenario.

The disposal of other waste types can lead to different functional diagrams. In the case of disposal of spent fuel, the waste matrix provided by the uranium oxide pellets is expected to degrade very slowly over a period that can last several hundreds of thousands of years. The corresponding functional diagram is given in Fig. 6.

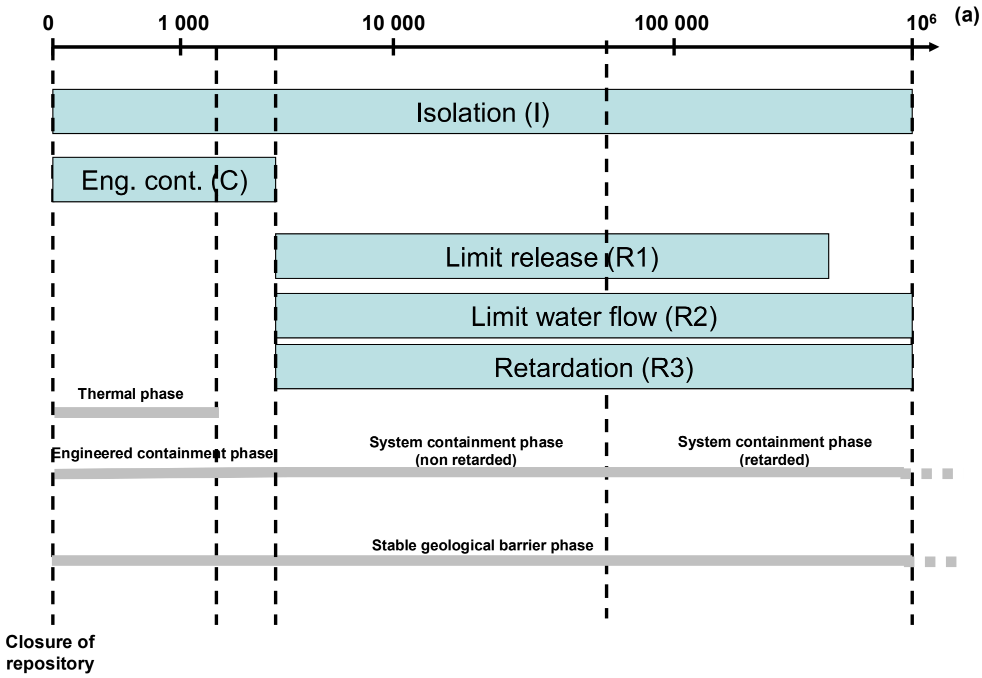


Fig. 6: Functional diagram of the repository system for the disposal of spent fuel in the case of the reference scenario.

In the case of disposal of intermediate level waste, no thermal phase will occur because the heat generated by those waste types is almost negligible. As a consequence, the use of a long-lived metallic overpack is not considered necessary. Further, the long-term stability of the waste matrices of the intermediate level waste types is often not well characterised; therefore, no credit is given to those waste matrices for the long-term safety. The corresponding functional diagram is given in Fig. 7.

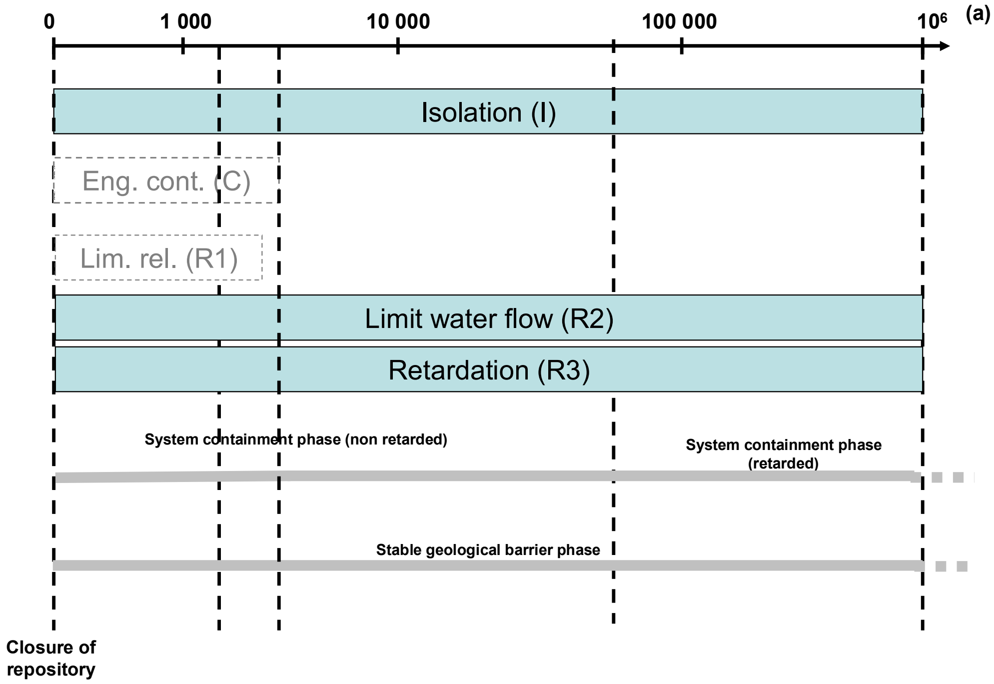


Fig. 7: Functional diagram of the repository system for the disposal of intermediate level waste in the case of the reference scenario.

6 Development of the safety statements

During the development of its long-term safety strategy for geological disposal (Dierckx, 2007) ONDRAF/NIRAS decided to develop a structured set (a tree) of safety statements that complements and underpins the set of safety functions. These safety statements are derived from the requirements on the disposal system as a whole, the various subsystems and the individual system components. During the development of the safety case the safety statements will initially be hypotheses (e.g. "the repository system should provide …") and they will develop into increasingly well-substantiated claims (e.g. "the repository system does or will provide …").

The top level (level 0) safety statement is "the repository system and its environment (should) conform(s) to regulatory standards and guidance concerning long-term safety via the safety functions that it performs over the required time frame". The underlying level 1 safety statements correspond to the three safety functions defined in Section 4.2, complemented with a statement concerning dilution in the environment (these four level 1 safety statements are directly substantiated by the assessment basis), a statement concerning the initial characteristics of the engineered barriers, and two safety statement referring to the results of the safety assessments and to the quality measures that have been applied in the assessment. The level 2 safety statements contain, among other, the sub-safety-functions.

An example of a limited (up to level 3) set of safety statements is given in Fig. 8. In the present set of safety statements developed for the SFC 1 up to 5 levels of safety statements have been worked out.

The safety statements are related to the long-term (or post-closure) safety of the repository. As the SFC 1 will also consider the feasibility of the considered geological repository, a set of feasibility statements is developed as well.

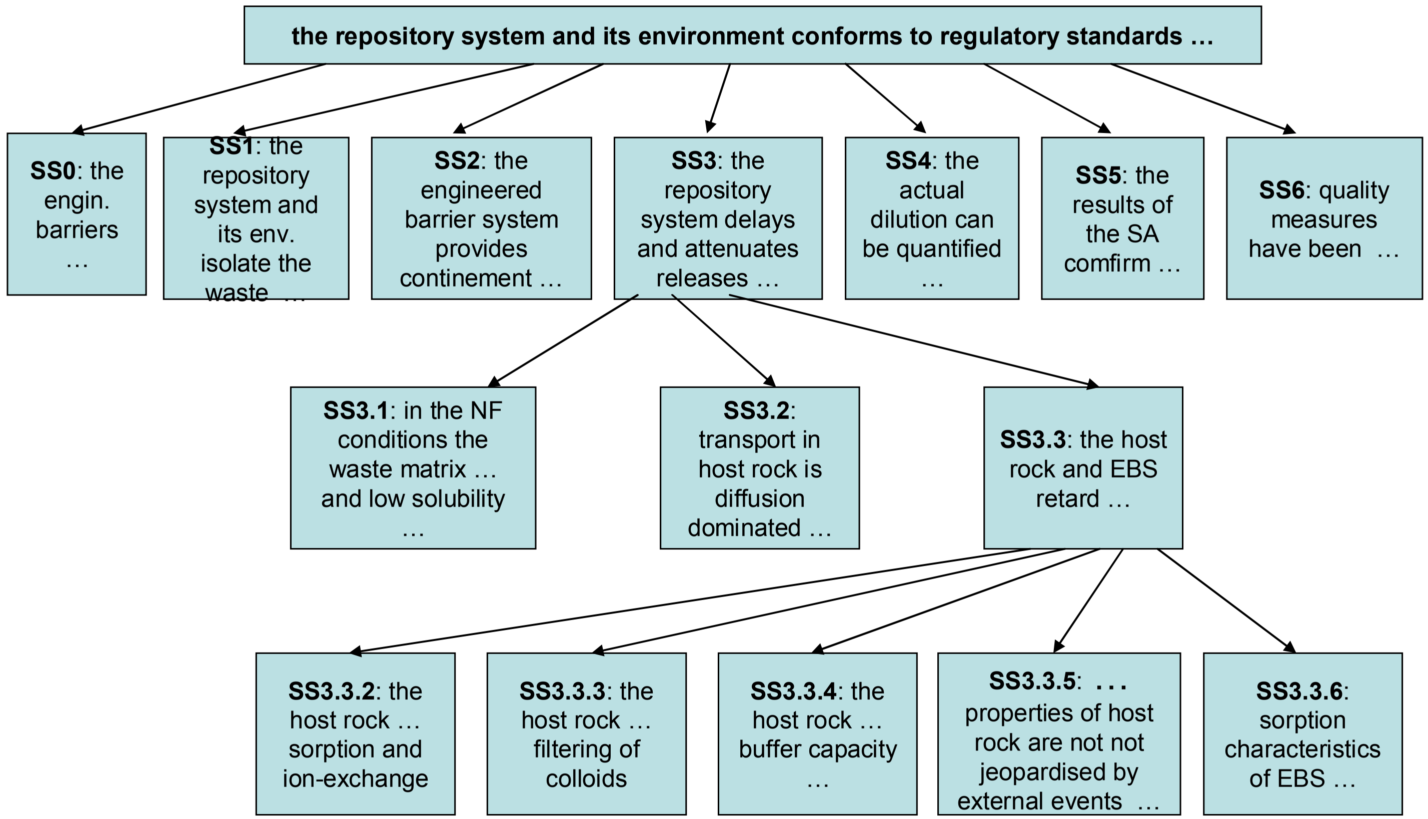


Fig. 8: An example of a tree of safety statements (limited up to level 3) developed for the SFC 1.

7 Uncertainty analysis

An important step in the proposed methodology is the identification of the uncertainties that can affect the substantiation of the safety statements.

For the SFC 1 it is planned that the research teams of the assessment basis will start this exercise by preparing a first list of uncertainties that can affect the safety statements during the second semester of 2008. This list will then be discussed between the research and safety assessment teams to check its completeness and consistency. As the planning of the scenario identification for SFC 1 is not compatible with the time schedule of WP3.1 of the PAMINA project, the application of the proposed methodology for scenario identification will necessarily be illustrated in the present report on the basis of a preliminary list of uncertainties.

The used preliminary list of identified uncertainties in the safety statements is given in Table 1. Table 1 contains 3 uncertainties that are due to an initial failure of one of the engineered barriers. The other identified uncertainties can affect one of the safety statements that are underpinning the safety functions or the dilution.

The next step is to analyse which identified uncertainties have the potential to affect a safety function. This analysis is also summarised in Table 1; the column "affected safety function" is filled in for the relevant identified uncertainties. In principle, these uncertainties can lead to an altered evolution scenario. The other uncertainties, the effect of which does not reach the level of a safety function, will be treated by considering an alternative modelling approach (conceptual model uncertainty) or a variant set of model parameters (parameter uncertainty).

Tab. 1: Preliminary list of identified uncertainties in the safety statements, the possible affected safety function, and proposed treatment (S: scenario uncertainty; M: model uncertainty; P: parameter uncertainty)

	Safety statement	Affected safety function	Treatment	
0	**Initial characteristics**			
0.1	**Defect of waste matrix**	R1	S	defect of waste matrix scenario
0.2	**Overpack defect**	C	S	initial overpack failure scenario
0.4	**Sealing failure**	R2, R3	S	poor sealing scenario
1	**Isolation (I)**			
1.1	**Reduce likelihood of human intrusion (I1)**			
1.1.2.1	area of limited valuable resources	I	S	human intrusion (HI scenarios will be treated separately)
1.1.2.2	effects on exploitation of water resources	I	S	deep well scenario
1.1.3	resilience of repository system in case of human intrusion	I	S	human intrusion (HI scenarios will be treated separately)
1.2	**Isolate to ensure stable conditions (I2)**			
1.2.2	erosion	I	S	low probability
1.2.3.1	effects of tectonic events (earthquakes, faulting, folding, uplift, subsidence) on disposal system	I	S	in discussion within assessment basis
1.2.3.3	effects of diapirism on disposal system	I	S	low probability
2	**Containment (C)**			
2.1.1	predictability of corrosion depth	C	S	early overpack failure
2.1.2.2	formation of passive layer		P	
2.1.2.3	no pitting corrosion		P	
2.1.2.4	no crevice corrosion		P	
3	**Delay and attenuate releases (R)**			
3.1	**Limit releases (R1)**			

3.1.1.1.2	HLW glass: dissolution rate		M, P	
3.1.1.1.3	HLW glass: influence of geochemical conditions		M, P	
3.1.1.1.4	HLW glass: specific surface area		P	
3.1.2.1	spent fuel: IRF		P	
3.1.2.2.1	spent fuel: matrix release rate		M, P	
3.1.2.2.3	spent fuel: influence of geochemical conditions on matrix release rate		M, P	
3.2	**Limit water flow (R2)**			
3.2.1.1	host rock structure: homogeneity		M, P	
3.2.1.3	host rock properties: variations with time		P	
3.2.3.5	gas transport	R2, R3	S	gas induced radionuclide transport
3.3	**Retardation (R3)**			
3.3.1	host rock properties: chemical conditions: low solubility		P	
3.3.2	host rock: sorption		P	
3.3.3	host rock: colloid filter		M	
3.3.4.1	extent alkaline plume		P	
3.3.4.2	extent oxidation front		P	
3.3.4.3	extent chemical disturbances due to temperature increase		P	
3.3.4.4	extent chemical disturbances due to waste products		M, P	
4	**Dilution and transfers in biosphere**			
4.1	dilution			
4.1.1	effect of climate changes on aquifers	D	S	variant of reference scenario
4.2	biosphere			
4.2.1	effect of climate changes on biosphere	D	S	stylised biospheres

8 Identification of altered evolution scenarios

Starting from the preliminary list of identified uncertainties in the safety statements (see Table 1), it is now examined which uncertainties can reach the level of the safety functions.

8.1 Uncertainties in initial characteristics

A defect of the waste matrix, e.g. strong fracturation of the matrix, affects sub-safety function R1 *limit release*. This sub-safety function is expected to contribute to the functioning of the repository system in the case of disposal of vitrified high-level waste and spent fuel. An initial defect of the waste matrix can be treated as an altered evolution scenario for these waste types. The functional diagram of the waste matrix defect scenario is given in Fig. 9. On the other hand, sub-safety function R1 does not appear in the safety concept for intermediate level waste, and consequently this scenario does not have to be considered for the intermediate level waste.

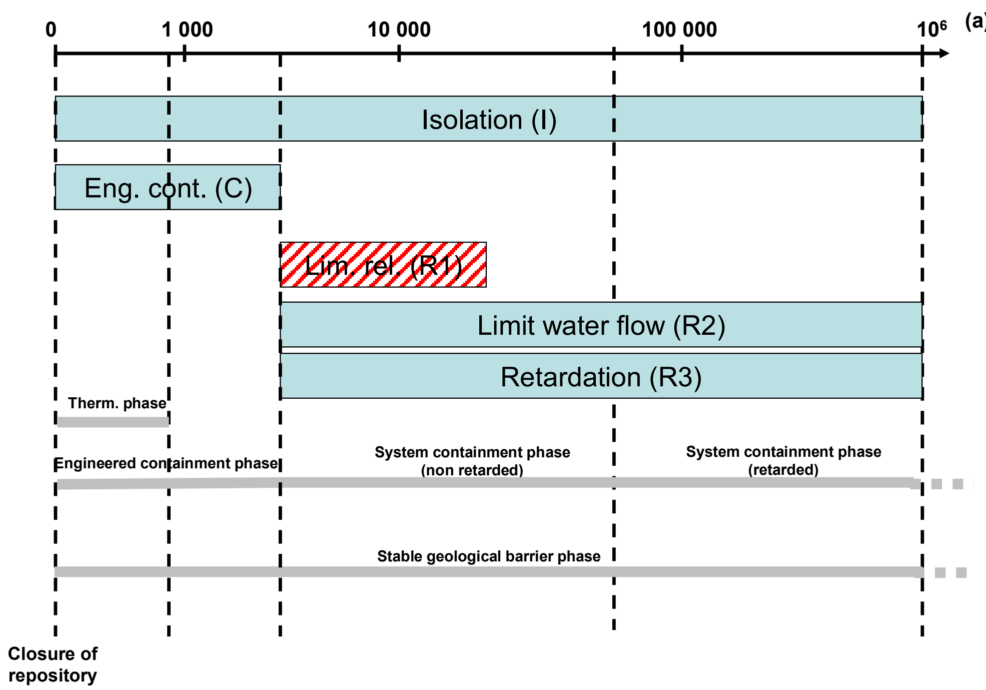


Fig. 9: Functional diagram of the repository system for the disposal of vitrified high-level waste in the case of the waste matrix defect scenario.

In the reference scenario for the disposal of vitrified high-level waste and spent fuel, a watertight overpack is expected to isolate the waste from groundwater at least during the thermal phase of the repository (probably much longer). An initial defect of the overpack makes that the safety function C *engineered containment* is not functioning, and as a consequence, the safety function R *delay and* *attenuate the releases* will have to function directly after waste emplacement. This case will be treated as an altered evolution scenario. The functional diagram of the repository system for the disposal of vitrified high-level waste in the case of the initial overpack defect scenario is given in Fig. 10.

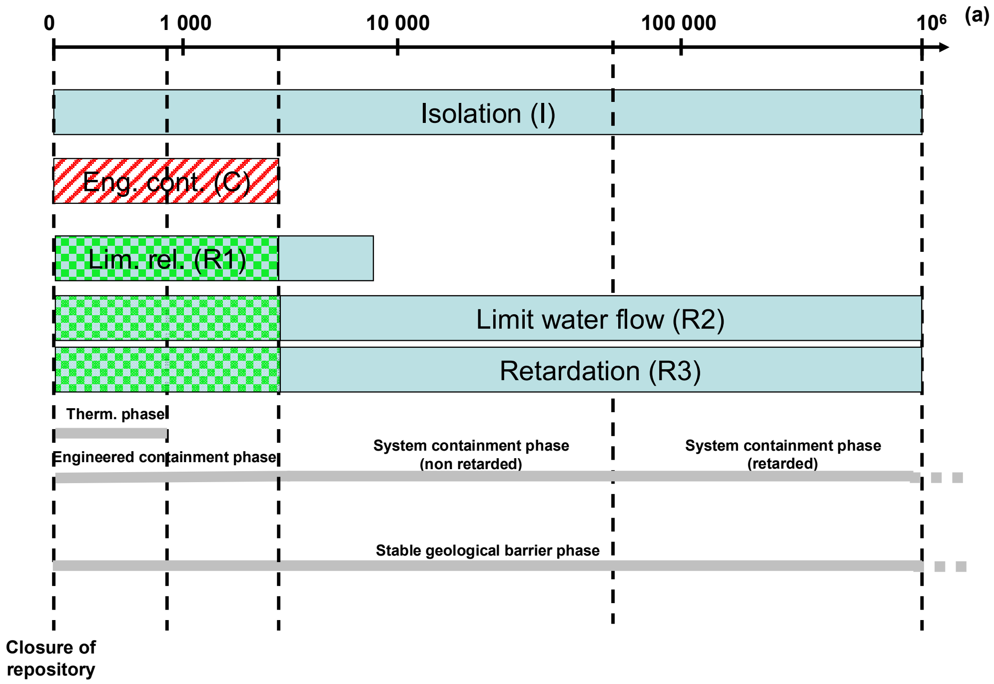


Fig. 10: Functional diagram of the repository system for the disposal of vitrified high-level waste in the case of the initial overpack defect scenario.

Various sealing failure scenarios can be taken into consideration. An extreme case is the abandonment of the unsealed repository. More realistic cases are that the sealing of the access shaft and the main galleries was less successful than designed, and that the hydraulic conductivity in the poorly sealed shaft and galleries is some orders of magnitude higher than the one of the Boom Clay. In this case the galleries might act as big filter that collects water from the host clay formation. In this case, sub-safety function R2 *limit water flow* is not available; sub-safety function R3 *retard contaminant migration* remains available in the host formation but only partially via the poorly sealed galleries and shaft pathway. This case has to be treated as an altered evolution scenario. The functional diagram of the repository system for the disposal of vitrified high-level waste in the case of the poor sealing scenario is given in Fig. 11.

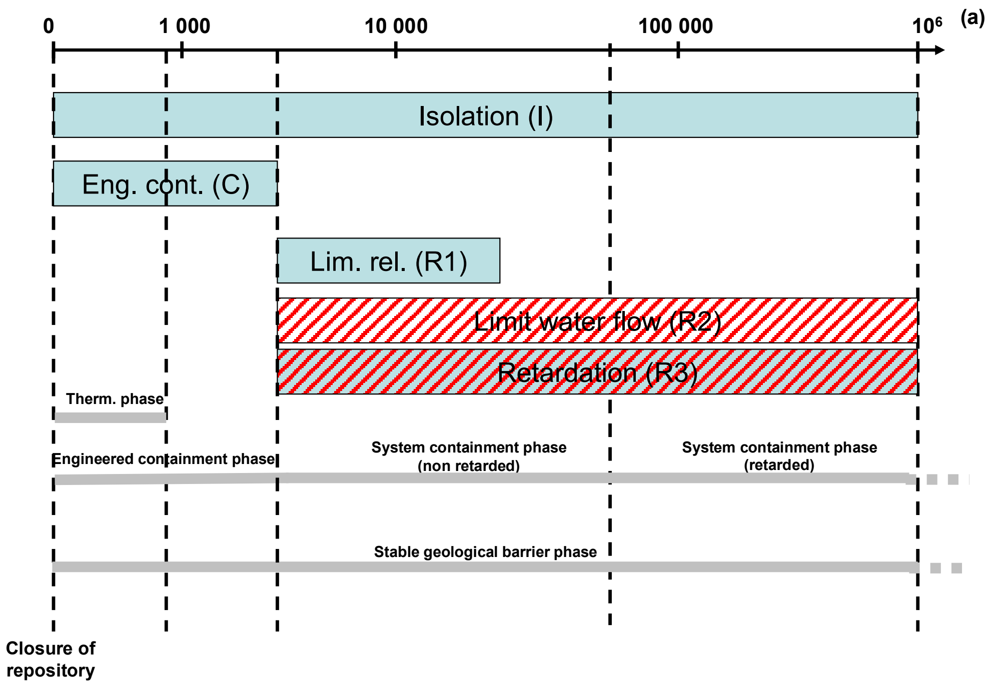


Fig. 11: Functional diagram of the repository system for the disposal of vitrified high-level waste in the case of the poor sealing scenario.

8.2 Uncertainties affecting the safety function isolation

For the safety function I *isolation*, two sub-safety functions are considered; these are: I1 *reduce likelihood of human intrusion* and I2 *isolate to ensure stable conditions*.

Human intrusion scenarios, affecting the *delay and attenuate* safety function and possibly even the *containment* safety function, will be treated separately in the safety assessment; therefore uncertainties that can lead to human intrusion scenarios will not be further handled in the present document.

Exploitation of water resources is not expected to affect one of the safety functions of the disposal system. Because the Neogene aquifer, which overlies the Boom Clay layer, is at present intensively used as a drinking water resource, a water well sunk in the Neogene aquifer is considered as a biosphere pathway in the reference scenario. However, the Under-Rupelian aquifer, this the aquifer which underlies the Boom Clay, is at present not used as a drinking water resource in the zone considered for geological disposal, because of its low pumping capacity. On the other hand, the deeper Lede-Brussel aquifer, which is separated by the about 30 m thick Asse Clay layer from the Under-Rupelian aquifer, is intensively used in the area south of the zone considered for geological disposal and sporadicly in the considered zone. This pathway, i.e. a pumping well sunk in the Lede-Brussel aquifer, will be considered as a variant case of the reference scenario.

Effects of erosion and diapirism are not considered as being relevant for the area considered for geological disposal. A study on the possible effects of tectonic events is going on, and it is expected that a decision on the treatment of those events in the SFC 1 will be taken at the end of 2009.

8.3 Uncertainties affecting the safety function containment

Uncertainties in the safety statements underpinning the containment safety function might lead to an early overpack failure. This scenario is probably less drastic than the initial overpack failure scenario already identified in § 8.2. The functional diagram of an early overpack failure is about identical to the one shown in Fig. 10. It is proposed to combine the initial and early overpack failure scenarios in one altered evolution scenario, for which a number of variant cases, e.g. the geometry of the hole in the overpack will depend on the mechanism that caused the hole, can be taken into consideration.

8.4 Uncertainties affecting the safety function delay and attenuate the releases

As for the safety function *delay and attenuate the releases* three sub-safety functions are considered, the uncertainties will also be discussed per sub-safety function.

8.4.1 Limit releases

For the case of disposal of vitrified high-level waste, the uncertainty in the longevity of the waste matrix can be treated as uncertainty in parameter values. For the case of disposal of spent fuel, uncertainty in the matrix degradation model can be considered as well; factors that can be treated as model uncertainty are, among others, the influence of the α-activity on the matrix degradation rate and the possible inhibiting effect of the presence of hydrogen generated by anaerobic corrosion of the steel of the overpack. An extreme case of uncertainty in the longevity of the waste matrix is the initial matrix failure scenario already identified in § 8.2.

Uncertainties in the chemical conditions prevailing in the near field can lead to uncertainties in the solubility limits of various radionuclides. These can be treated as parameter uncertainty.

8.4.2 Limit water flow

Uncertainties concerning host rock structure and characteristics can be treated respectively as model and parameter uncertainty.

For disposal of vitrified high-level waste and spent fuel, it is expected that in the case of the reference scenario the hydrogen generated by anaerobic corrosion of the carbon steel overpack can be evacuated by diffusive transport into the host clay formation. Higher hydrogen generation rates might lead to disruptions of gas into the host formation. If this occurs after perforation of the overpack, gas disruptions might lead to gas mediated or possibly to gas induced transport of radionuclides. These unexpected gas phenomena can be treated in an altered evolution scenario.

In the case of disposal of intermediate level waste, e.g. compacted hulls and end-pieces, the specific surface area of metal present in 1 metre of a disposal gallery is much higher than in the case of high-level waste disposal; as a consequence, the hydrogen generation rate can be much higher than in the case of high-level waste disposal, and (controlled via galleries and shafts or uncontrolled into the host formation, depending on the repository concept retained for the disposal on intermediate level waste) gas disruptions may have to be considered within the reference scenario.

8.4.3 Retardation

The uncertainties that are identified for the sub-function retardation can be treated as parameter or model uncertainty, and consequently they do not lead to an altered evolution scenario.

8.5 Uncertainties affecting the dilution

Climate changes are not expected to have a considerable effect on the safety functions themselves; on the other hand they can have a drastic impact on the dilution in the aquifers surrounding the host formation.

In the beginning of the nineties, safety assessments of geological disposal assumed that future climates will be driven by Milankovitch's orbital theory. This theory forecasts a moderate glaciation in about 25,000 years and a more severe glaciation, comparable with the Weichselian glaciation, in about 55,000 years (Berger et al., 1991). However, in recent years several climate experts claim that the greenhouse effect will be long-lasting and that no glaciations have to be expected in the next one to two hundreds of thousands of years (Berger and Loutre, 2000).

It is proposed to treat uncertainties in the evolution of the climate as variants of the reference scenario (because the safety functions are not affected). At least two variants will have to be considered: one mainly based on Milankovitch's orbital theory and another in which a strong greenhouse effect is taken into account.

The biosphere is also strongly influenced by the climate. Therefore, a number of stylized reference biospheres, corresponding to the main climate types that can be expected to occur in the zone considered for geological disposal, will be developed.

9 Conclusions and prospects

This report describes a method for systematic scenario identification and illustrates its application to the case of geological disposal of radioactive waste in the Boom Clay formation in NE Belgium. The proposed method starts from a list of uncertainties in the safety statements. It is examined which uncertainty can affect a safety function. Four altered evolution scenarios have been identified. These are:

- initial matrix defect scenario;
- initial overpack defect scenario;
- poor sealing scenario;
- gas mediated or induced radionuclide transport scenario.

Further, a number of variant cases of the reference scenario were also identified.

The work reported in the present document is just a first application of the proposed methodology. It will be further developed and tested during the next months in the framework of the safety assessments for the Safety and Feasibility Case 1 of ONDRAF/NIRAS, which is scheduled to be finalised in 2013.

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Part 1.2: Third Level Decomposition of Safety Functions

(Prepared by A. Vokál, NRI, Czech Republic)

Abstract

In this report the approach for systematic derivation of scenarios for evaluation of safety of the concept of disposal of spent fuel assemblies in carbons steel canisters surrounded by bentonite in granite structure is summarised. The main principle of the approach is to define first safety functions of the disposal system, subsystems and its components in top-down, hierarchical arrangement and then to analyse these safety functions in respect of all possible interactions with surrounding systems, subsystems and components. The safety functions were divided into two categories. Safety functions strictly related to the main objective of the disposal system to limit the releases to biosphere by containment and delay so that the exposure of people is lower than regulatory limits belong to the first category. The second category of safety functions covers all the assumptions under which the direct safety functions must work. Five basic scenarios, which include the normal evolution scenario, scenarios initiated by an event or process from interactions of the disposal system with external systems (environment and development and construction activities) were derived. These scenarios are evaluated by a systematic analysis of the effect of initiating events or processes on safety functions of subsystems and components in a bottom-up approach.

1 Introduction

This report is an internal document prepared by NRI in the framework of EU project "PAMINA". It serves for developing a methodology of derivation of scenarios on the basis of safety functions for concept o disposal of spent fuel assemblies in carbon steel canisters in granite host rock. This approach is based on systems engineering approach, which integrates the total engineering effort to meet cost, schedule, and technical performance objectives in managing complex projects. It usually includes the following steps (in logical, not temporal order):

- Objective (problem) definition
- value system design
- function analysis
- systems synthesis
- systems analysis
- decomposition
- description.

Performance assessment disposal system is usually segregated into the following steps:

- Identification of all relevant features, events and processes (FEPs), often termed factors; their structuring and ranking (screening); the selection of those factors that should be included in quantitative analyses; and scenario development
- Development, verification and validation of models and computer codes
- Calculation of the consequences of each scenario
- Estimation of the uncertainties in the results and identification of the parameters and assumptions that are of most importance
- Comparison of the results with the appropriate standard or criteria.

All steps included in performance assessment steps are also included in systems engineering. The identification of FEPs and the scenario development are covered by function analysis; the development, verification and validation of models and codes, calculation of consequences, estimation of uncertainties and comparison of the results with appropriate criteria are common steps of systems analysis.

The development of safety functions in Czech DGR program was based on systems engineered approach called FRAT (Functions, Requirements, Answers and Tests) developed in USA [1]. The following steps characterize this approach:

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.

2. 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 that 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 (process) 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.

As can be seen from Fig. 1, upper level FRAT data provide scope for next level, lower level must roll-up and map to upper level and each level of FRAT establishes a baseline.

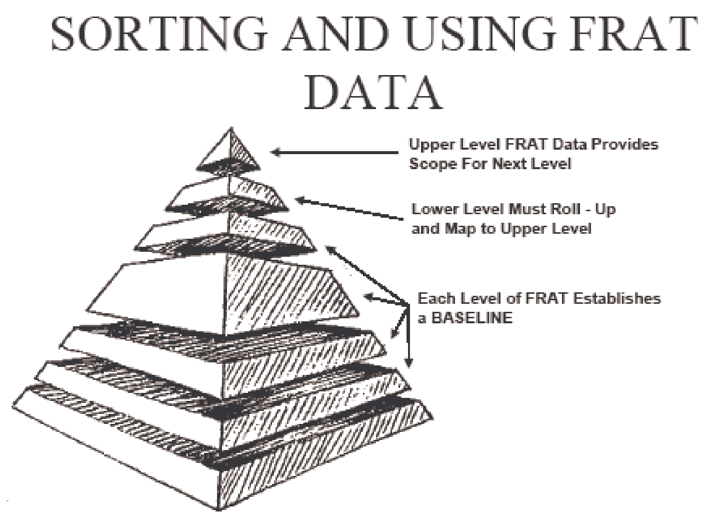

Fig. 1: Sorting and using FRAT data [1]

This system can be used both for developing new systems fulfilling some functions and for evaluating already proposed system. In this approach, first, **F**unctions (safety functions) are identified for a given **A**lterantive, the functions are then connected with **R**equirements in quantitative form. Scenarios and following consequence calculations are here understood as **T**ests serving for evaluation of safety of proposed disposal system and its components fulfilling some functions.

2 Objectives of disposal system

The main objectives of underground disposal system can be in agreement with IAEA Safety Standards [2] formulated as follows:

To isolate high level wastes from the human environment and to ensure the long-term radiological protection of humans and the environment.

It is believed that this objective can be fulfilled by emplacing of spent fuel assemblies and high level waste in carbon steel canisters surrounded by compacted bentonite in granite host rock (disposal system). The disposal system interacts with surrounding systems which affect their safety functions (Fig. 2):

- Environment system
- DGR development and construction activities system

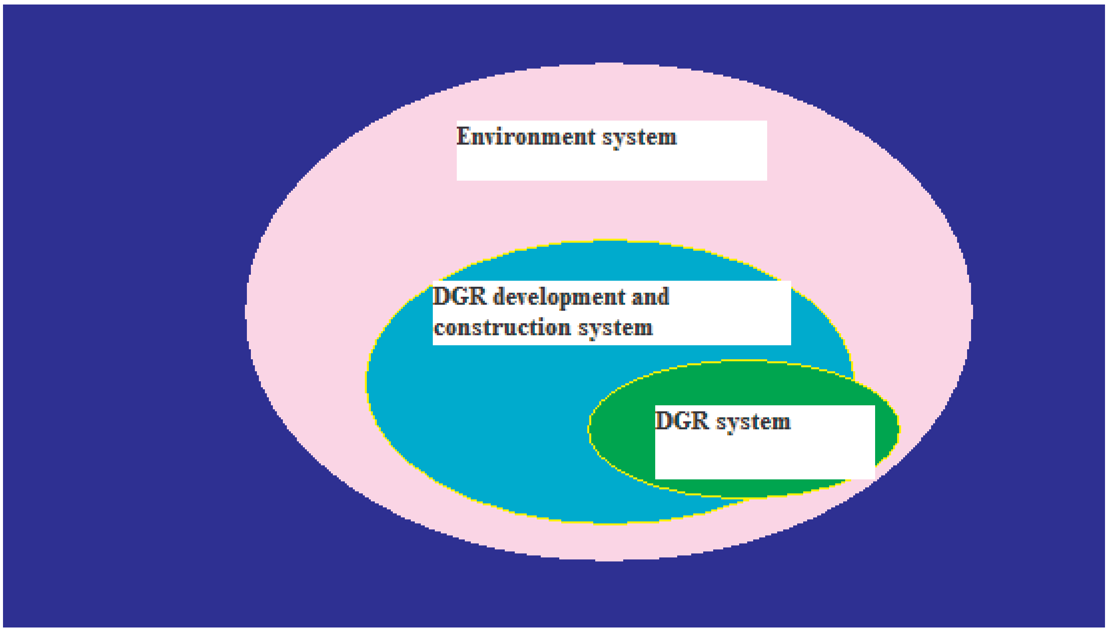

Fig. 2: DGR and surrounding systems

This disposal system in interaction with surrounding systems should guaranty that the main objectives of underground disposal will be fulfilled. The main product in our approach is disposal system, which is defined as the combination of engineered and natural barriers that isolate radioactive waste when disposed in geological formations. Environment (site) and development and construction activities create important constraints for disposal system, which can be also formulated in the way of safety functions and requirements.

3 The value system definition - the basic legislative requirements

The legislative requirements form the initial point for definition of quantitative safety requirements and consequently performance objectives of disposal system. The waste management legislation in most countries, as is the case also with the Czech Republic, follows the recommendations of IAEA (2). According to safety principles, formulated in these standards, 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. Gradual processes are considered to include all evolutionary processes affecting the disposal. Disruptive processes are those processes that occur as random events and may have a disruptive effect on the repository and its environment.

According to the Czech Atomic Act [3] , approved in January 1997, and relevant regulations all practices 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 and without this definition it is difficult to develop a design of an engineered barrier system. The difficulty of this vague definition has been solved with the aid of methods that enable to identify and structure possible features, events and processes (FEPs) in such way that no important FEPs would be neglected. The main objective of the methods, such as the Process Influence Diagram (DIP) or Rock Engineered System (RES) or Master Directed Diagrams (MDD) [4-6] is to structure features, processes and events (FEPs) in such a way that it is visible that no important factor was neglected or forgotten so that all disposal system functions and requirements have been defined.

The guidance level of exposure of 250 μSv, which is considered as sufficient for the evidence of reasonably achievable level of radiation protection for the annual effective dose for a critical group of population, is defined in Czech legislation. 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. It can be expected that if this limit for people is not exceeded that also the limit for the environment will not be exceeded. All possible ways of exposure of people must be taken into account.

4 Derivation of safety functions for concept carbon steel/ bentonite/ granite

4.1 First level safety function

The history and current utilization of safety functions in performance assessment of deep geological repositories was summarized in one of the items of WP1 of PAMINA project. 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 [7] Belgian [8] and French [9] radioactive waste management programmes.

It was concluded that three main categories of safety functions can be distinguished:

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

The importance that is given to a specific safety function strongly depends on the host formation. In the 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.

The first step in the hierarchical systems engineering approach used in this work is to define firstly, a top safety function. The top safety function of the proposed disposal system was formulated as follows:

F.1 Limit releases of radionuclides to the environment by containment and delay

This function covers at this level both "containment" and "limit and delay" function identified in WP1 and is directly related to the main objective of a disposal system to isolate high level wastes from the human environment and to ensure the long-term radiological protection of humans and the environment.

As will be discussed later stability/isolation safety function is in this approach considered as a supporting safety function and is allocated both on the environment, development and construction activities and disposal system itself.

4.2 First level supporting safety functions

A disposal system will work only under some assumptions concerning the environment and development and construction activities and also disposal system itself. These assumptions can be transferred to safety functions allocated on the environment (site) and development and construction activities and disposal system. The three supporting top safety functions were formulated as follows:

FS1: Provide stability against the effect of the environment and human errors – Disposal system itself

FS2: Provide favourable conditions for DGR – Environment system

FS3: Ensure minimum human errors through QA system - DGR development and construction system

The supporting safety function FS1 can be identified with "stability/isolation" safety function formulated in WP1 of PAMINA project.

4.3 Second level safety functions

At the second level, the proposed disposal system was decomposed into 3 subsystems:

- Repository (Engineered Barrier System)
- Host rock
- Biosphere

There is no direct safety function for biosphere due to great susceptibility of biosphere to changes in horizon of millions of years.

The main direct safety functions for repository and host rock were formulated as follows:

F1.1 Limit releases of radionuclides to geosphere by containment and delay – repository

F1.2 Limit transport of radionuclides to biosphere by delay, dispersion and dilution – granite host rock

At this level of the system decomposition, repository can be considered as a black box of certain dimensions covering both the waste and all the engineered materials introduced in the host rock system by man. Boundary between the repository and host rock are at the interface between buffer/backfill and host rock. Boundary between host rock and biosphere can be defined as part of host rock, which can be easily changed or affected by surface processes being it natural or human.

4.4 Second level supporting safety functions

The supporting safety (FS1 – FS3) can be further decomposed through detailed analyses of Features, Events and Processes according to structure of international NEA/OECD database of FEPs [10] e.g. as follows:

Disposal system

FS1.1 Provide favourable, stable conditions for engineered barrier system – host rock

FS1.2 Provide favourable, stable conditions for host rock and repository itself - repository

Environment

FS2.1 Provide favourable geological processes and events – processes arising from the wider geological setting and long-term processes

FS2.2 Provide favourable climatic processes and events – processes related to global climate change and consequent regional effects;

FS2.3 Provide favourable environment for future human actions – human actions and regional practices in the post-closure period that can potentially affect the performance of the engineered and/or geological barriers (not intrusive actions)

Development and construction activities

FS3.1Provide decisions on design and waste allocation, and operations and closure and issues favourably affecting initial state of components (Repository issues)

FS3.2Provide favourable conditions for site investigation

FS3.3 Provide operations and closure activities favourably affecting initial state of components

Only supporting safety functions related to disposal system (*FS1.1 and FS1.2*) will be discussed in this work.

4.5 Third level safety functions

4.5.1 Repository safety functions

At the third level of disposal system decomposition, a repository (engineered barrier system) for concept carbon steel/ bentonite/ granite was divided into:

- Spent fuel assemblies (matrix, structure materials, instant release fraction, etc.)
- Carbon steel waste package
- Buffer (backfill)
- Other engineered structures (backfill, seals, etc.)

Direct safety functions of repository in relation to safety function F1.1 were formulated as shown in Table 1. No direct safety function is allocated on other engineered structures, which have only direct function in operating period of a repository and supporting functions in period after repository closure. The exception can be backfill, which could have some direct safety function depending on concrete disposal concept.

Tab. 1: Direct safety functions of EBS

ID function	Function description	Allocated to
F1.1.1	Limit leaching of radionuclides from spent fuel assemblies	Spent fuel assemblies
F1.1.2	Isolate radionuclides as long as possible	Canisters
F1.1.3	Delay release of radionuclides to geosphere	Buffer

Spent fuel assemblies, canisters or buffer at this level represent all individual components in a repository.

4.5.2 Host rock third level safety functions

Granite host rock was divided into the following components:

- Intact host rock surrounding disposal units including EDZ (IHR)
- Hydraulically active zone (HAZ)
- Sealed structure of host rock (SS)

- Excavated Damage (Disturbed) Zone (EDZ)

In reality the properties of undisturbed granite host rock zone surrounding disposal units will be at variance significantly one disposal unit from another. Contrary to host rock like clay it is thus very difficult to average properties of all disposal units due to high heterogeneity of granite host rock. It is probable that some of disposal units will have to be located in fractured areas, which will be less favourable for fulfilling safety functions than other. But at this level of system decomposition and without concrete host rock properties it is difficult to differentiate all disposal units and therefore some more precise differentiation will have to be carried out only after detailed characterization of selected site. Due to this large variation of properties, in our opinion there is no sense at this level to differentiate between intact rock and excavated damaged or disturbed zone (EDZ). At this level it is difficult to differentiate the properties of EDZ from the properties of other host rock surrounding disposal units. So that safety function for EDZ was not defined. It is also expected [11] that in granite important EDZ is not formed if good engineering and adequate QA practices are applied.

The primary safety functions of host rock components are formulated in Tab. 2.

Tab. 2: Direct safety functions of host rock

ID function	Title	Allocated to
F1.2.1	Delay transport of radionuclides to hydraulically active zone by very low advection, matrix diffusion, sorption and dispersion	IHR
F1.2.2	Delay transport of radionuclides to the environment by low advection, matrix diffusion, sorption, dispersion and dilution	HAZ
F1.2.3	Delay transport of radionuclides through sealed structure	SS

4.6 Third level supporting safety functions

Repository and host rock has a number of supporting safety functions, which must be established to protect other both direct and supporting safety functions both of repository components itself and host rock. The derivation of these supporting safety functions was carried out on the basis of OECD/NEA FEPs database [10] – Disposal system domain. But contrary to OECD/NEA database only the following processes and events were at this level taken into account. The factors like inventory, radionuclide and other material, waste form materials and other characteristics or container materials and characteristics and further are taken as items for which safety functions will be later derived.

- Mechanical processes and conditions
- Hydraulic/hydrogeological processes and conditions
- Thermal processes and conditions

- Chemical/geochemical processes and conditions
- Biological/biochemical processes and conditions
- Gas sources and effects
- Radiation effects
- Nuclear criticality

4.6.1 Host rock supporting safety functions

It is well known that one of the most important safety functions of host rock is to provide favourable conditions for engineered barrier system (or to protect engineered barrier system) – *FS1.1*. This function was divided into third level safety functions according to FEPs analysis. These third level supporting safety functions allocated on host rock are given in Table 3.

Tab. 3: Supporting safety functions for host rock

ID	Function Description
FS1.1.1	Provide favourable mechanical conditions for disposal system safety functions
FS1.1.2	Provide favourable chemical conditions for disposal system safety functions
FS1.1.3	Provide favourable thermal conditions for disposal system safety functions
FS1.1.4	Provide favourable hydrogeological conditions for disposal system safety functions
FS1.1.5	Provide transfer of gases originating in disposal system safety functions
FS1.1.6	Provide favourable microbial activities
FS1.1.7	Provide conditions protecting against possibility of critical nuclear reactions in host rock or repository itself safety functions

4.6.2 Repository supporting safety functions

Repository itself must provide favourable conditions for safety functions of repository and host rock. The structuring of supporting safety function is the same as for host rock (Table 4).

Tab. 4: Supporting safety functions of repository

ID function	Function description
FS1.2.1	Prevent mechanical failure
FS1.2.2	Provide low flux of water through repository
FS1.2.3	Provide favourable chemical conditions for disposal system safety functions
FS1.2.4	Provide favourable thermal properties for heat transfer from a repository
FS1.2.5	Provide favourable microbiological properties for repository safety functions and host rock itself
FS1.2.6	Limit amount of gases originating in a repository and enable transfer of gases through repository
FS1.2.7	Provide conditions unfavourable for critical reactions and host rock itself

4.7 Fourth level safety functions

4.7.1 Spent fuel assemblies safety functions

Spent fuel assembly safety function F.1.1.1 Limit leaching of radionuclides from spent fuel assemblies is allocated on the following spent fuel assembly components:

- Spent fuel matrix
- Cladding
- Other structure materials
- Gap and grain boundaries (IRF fraction)

It is evident that it is not possible “to design” the safety functions for spent fuel assemblies, because they are given by conditions of irradiation in nuclear reactors, but it is advantageous to formulate safety functions for these components as well as to set some limits and criteria for spent fuel assemblies accepted to repository (which can be transferred later to waste acceptance criteria – WAC for repository operators and waste specifications for waste producers). The safety functions for spent fuel assemblies are formulated in the Table 5 so that any waste package exceeding some criteria related to these functions will not be accepted to a repository without additional safety assessment.

Tab. 5: Spent fuel assemblies safety functions

ID	Title	Allocated to
F1.1.1.1	Limit leaching of radionuclides from spent fuel matrix	Spent fuel matrix
F1.1.1.2	Limit leaching of radionuclides from cladding	Cladding
F1.1.1.3	Limit leaching of radionuclides from spent fuel structure materials	Structure materials
F1.1.1.3	Limit leaching of radionuclides from gap and grain boundaries	Gap and grain boundaries

Leaching of radionuclides from spent fuel assembly components will be quantified by fraction of radionuclides released from waste, i.e. by radionuclides contained in the components and by their leaching rate after their contact with water.

4.7.2 Carbon steel canister (waste package) safety functions

The direct safety function of canisters (waste packages) is to isolate radionuclides practically completely as long as possible so that no radionuclide can escape to bentonite and geosphere for some time. This safety function is very important especially for concept in fractured granite host rock. Carbon steel based canisters, which have been proposed in the Czech concept, are shown in Fig. 3. The canister is composed of three layers, which protect spent fuel assemblies against contact with water. The first barrier is a layer made of corrosion resistant Ni alloy (3 – 6 mm), the second barrier is a thick layer of carbon steel (80 mm) and the third one is a thin (5 mm) wall of inner canister made of stainless steel. Spent fuel assemblies are emplaced in special baskets, which hold spent fuel assemblies in place and prevent initiation of critical reactions in canisters.

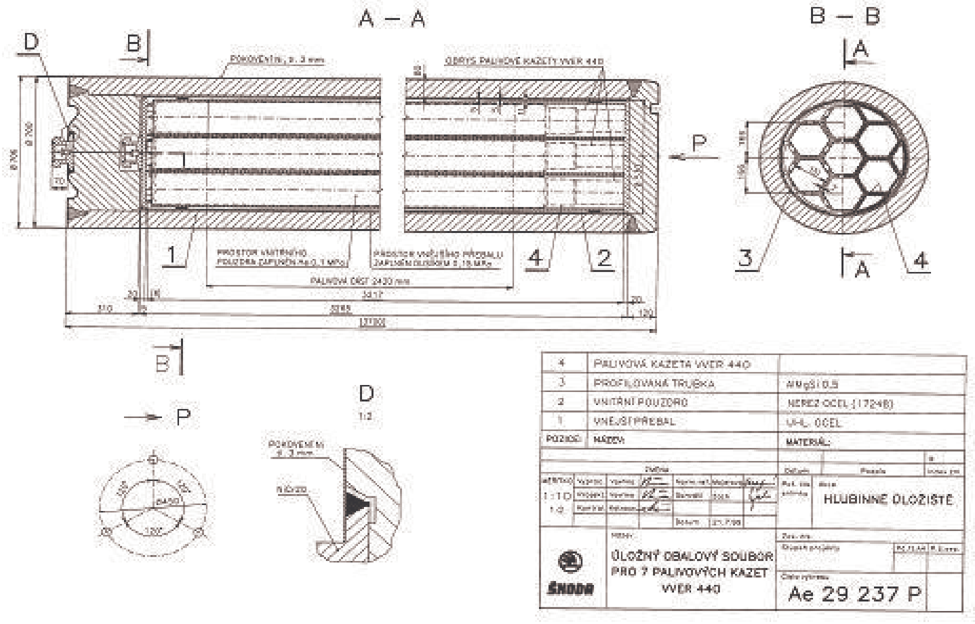

Fig. 3: Scheme of disposal canister for spent fuel assemblies from WWER 440 reactors

Another important barrier preventing direct contact of spent fuel assemblies with water is spent fuel cladding made of thermodynamically stable zirconium alloy.

Under reducing conditions the corrosion rate of Ni-alloys is very low (< 0.1 μm/rok), but pitting or crevice corrosion leading to the formation of localized defects cannot be excluded. We estimate that the lifetime of this layer can prolong for thousands of years. So that the time of contact of carbon steel with water can hardly be earlier than after several tens of thousand of years. After this period corrosion of carbon steels will start. Long-term steady state corrosion of carbon steels in reducing conditions is usually not higher than 0.1 - 3 μm/yr depending on temperature and other variables. This means that a corrosion allowance layer of 60 mm from 80 mm thick wall of the canisters will corrode out first in further 30 000 years.. After this time, some of canisters could be destroyed with possible formation of fractures due to hydrostatic pressure and pressure of swelling bentonite. But still spent fuel assemblies will not be exposed to water because of stainless steel inner canisters, which further protect fuel assemblies against contact with water. The corrosion rate of stainless steel canister is very low in reducing conditions and it will require further thousands of year to corrode out. Only now radionuclides from surface contamination of spent fuel assemblies structure materials can be released, but other radionuclides included in spent fuel assemblies are still protected by a highly corrosion resistant layer of cladding.

It seems that under normal evolution scenario radionuclides cannot be released from canisters for possibly hundred thousands of years. This is, however, an estimate, which will have to be proved by some experimental data and model calculations. All mentioned layers contribute to the fulfilling of canister safety function (F1.1.2). These daughter safety functions of canister components were formulated as follows (Table 6):

Tab. 6: Waste package components safety functions

ID	Title	Related to
F.1.1.2.1	Protect carbon steel against contact with water especially in aerobic period	Ni alloy
F.1.1.2.2	Protect stainless steel inner canister against contact with water in anaerobic period	Carbon steel
F.1.1.2.3	Protect spent fuel assemblies against contact with water	Stainless steel
F.1.1.2.4*	Protect spent fuel rods against contact with water	Cladding

* Cladding is here considered as part of waste packages protecting spent fuels matrix from contact with water

It can be expected if these components will meet their safety functions that safety function of canister will be also met if some other supporting safety functions allocated on canister itself and surrounding components are fulfilled.

4.7.3 Direct buffer (and backfill) safety functions

Direct safety function of buffer is "to delay release of radionuclides to geosphere". This function is important primarily for mobile radionuclides, which could get quickly from failed canister located in unfavourable conditions to the environment. Radionuclides are delayed by low solubility, sorption and slow diffusion in buffer. The important factor for delay of radionuclides in buffer is also the thickness of bentonite layer. The safety functions of buffer were divided into the safety functions given in Table 7.

Tab. 7: Buffer direct safety functions

ID	Title
F.1.1.3.1	Provide low solubility of radionuclides
F.1.1.3.2	Provide transport by diffusion only
F.1.1.3.3	Provide high values of sorption coefficients
F.1.1.3.4	Provide sufficient thickness for delay by diffusion

4.7.4 Intact host rock safety functions

An "intact host rock zone" can be defined as host rock which does not contain large hydraulically active fracture zones with fast paths to the environment. The distance from the major faults is also an important barrier protecting repository against negative seismo-tectonic changes. From this reason it is considered that this part of host rock is the most important barrier for delay of transport of radionuclides to the environment. The important parameters affecting transport of radionuclides in this zone are long travel time to hydraulically active zone, porewater chemistry ensuring low solubility (low Eh, low amount of colloids, etc) and high retardation of radionuclides due to sorption and diffusion to rock matrix. The direct safety functions of this zone were formulated as given in Table 8.

Tab. 8: Direct safety functions of undisturbed host rock

ID	Title
F1.2.1.1	Provide long time of transport to hydraulically active zone
F1.2.1.2	Provide high values of retardation
F1.2.1.3	Provide low values of radionuclides solubilities

4.7.5 Hydraulically active zone

Safety functions of hydraulically active zone are in principle similar to undisturbed zone. This host rock zone must also delay transport of radionuclides to biosphere by long time of travel, by high sorption and low solubility. In addition to hydraulically active zone would contribute to lower impact of repository on the environment by dilution and dispersion. (Dilution and dispersion will have certainly an important impact also on intact host rock, but we suppose that it is very difficult to evaluate it so no such function was allocated on intact host rock zone). The safety functions of hydraulically active zone are given in Table 9.

Tab. 9: Safety functions of hydraulically active zone

ID	Title
F1.2.1.1	Provide long time of transport to biosphere
F1.2.1.2	Provide high values of retardation
F1.2.1.3	Provide low values of radionuclides solubilities
F1.2.1.4	Provide dilution and dispersion

4.7.6 Sealed structures

Safety functions for sealed structures are the same as for intact host rock zone. In our opinion it is important to define safety function for sealed fracture, because this enables to formulate requirements and specification for seals.

4.8 Fourth level of supporting safety functions

At fourth level of system decomposition there is a very large number of supporting safety functions. Detailed analyses exceed the framework of this work. It will be based on systematic analyses of interactions between third level components using expert elicitation approach.

Only some examples of fourth level supporting safety functions are given below:

4.8.1 FS1.1.1 "provide favourable mechanical conditions for disposal system"

Host rock can influence a repository by isostatic load and by shear load. This load must be limited both in respect to bentonite buffer and canister. Fourth level supporting safety functions can be thus formulates as follows:

FS1.1.1.1 Limit isostatic pressure on disposal units

FS.1.1.1.2 Limit shear at deposition holes to avoid canister failure due to rock shear

4.8.2 FS1.1.2 Provide favourable chemical conditions for disposal system safety functions

Host rock must contribute to safety functions of engineered barriers by providing suitable chemical conditions. These chemical conditions must be related in respect to all EB components: waste form, canister and buffer. Examples of daughter supporting safety functions are as follows:

FS1.1.2.1 Provide low redox condition for low corrosion rate of canister and low solubility of radionuclides

FS1.1.2.2 Provide water composition (limited concentration of detrimental agents) to avoid corrosion rate of canister and waste form and degradation of bentonite

FS1.1.2.3 Provide ionic strength to avoid buffer erosion (minimum ionic strength)

FS1.1.2.4 Provide limited alkalinity to avoid dissolution of buffer

FS1.1.2.5 Provide limited salinity to avoid detrimental effect on buffer swelling

4.8.3 FS1.1.3 Provide favourable thermal conditions for disposal system safety functions

FS1.1.3.1 Provide transfer of heat from repository to avoid degradation of barriers

4.8.4 FS1.1.4 Provide favourable hydrogeological conditions for disposal system safety functions

FS1.1.4.1 Provide limited flux around disposal units to prevent fast degradation of barriers and erosion of bentonite

4.8.5 FS.1.1.5 Provide favourable conditions for gas transport

FS1.1.5.1 Provide favourable conditions for hydrogen transport generated during anaerobic corrosion of canisters before their failure

FS1.1.5.1 Provide favourable conditions for hydrogen transport generated during anaerobic corrosion of canisters before after their failure

4.8.6 FS1.1.6 Provide favourable microbial conditions

FS1.1.6.1 Provide favourable microbial conditions for low corrosion of canister

FS1.1.6.2 Provide favourable microbial conditions for low corrosion of waste form

Some more examples of supporting safety function are given in Table 10.

Tab. 10: Examples of further supporting safety functions for carbon steel/ bentonite/ granite concept

ID function	Title	Related to	Remark
FS1.2.1.1	Prevent mechanical failure of canisters by seismo-tectonic changes	Bentonite	Plastic behaviour of bentonite buffer seismo-tectonic stress in host rock
FS1.2.1.2	Prevent mechanical failure of canisters by swelling pressure of bentonite	Bentonite	Bentonite swelling pressure could leads to premature failure of canister
FS1.2.1.3	Protect mechanical failure of canisters by inner filler	Inner structure of canister	Inner structure of canister can protect canister against mechanical failure
FS1.2.1.4	Protect mechanical strength of canister by corrosion allowance layer of overpack	Corrosion allowance layer of overpack	Corrosion allowance layer prevent premature mechanical failure of canisters

5 Safety requirements

In agreement with FRAT approach safety functions will be provided by requirements in a quantitative form. These requirements does not mean in the beginning of the project development that if they are not fulfilled that the concept is wrong. They have tentative character and can be changed in agreement with increasing knowledge of the system. It can be expected that in a final stage of disposal system development the requirement for selected engineered barriers will be transferred to exact specifications of barriers.

The requirements for direct safety functions should be strictly related to the main quantitative requirement for a disposal system, namely to annual effective dose (250 μSv/yr). They can be derived with help of the following simplified analytical equations (in-growth of radionuclides in decay chains is not included):

Annual effective dose E_i for a given scenario is calculated as a summation of effective doses from individual radionuclides E_j through their ingestion or exposure by different means (drinking water, foods, etc).

$$E_i = \sum_j^n E_j \qquad (1)$$

The E_j is calculated from the concentration of radionuclides in the flowing water at a discharge point into the biosphere ($C_{bio,j}$) or in surface soil using Biosphere Dose Conversion Factors (BCDFs):

$$E_j = C_{bio,j} \cdot BDCF_j \qquad (2)$$

Concentration of radionuclides is related to concentration of radionuclides at the interface repository/geosphere ($C_{geo,j}$):

$$C_{bio,j}(t) = \frac{C_{geo,j}(t) \cdot e^{-\lambda_j \cdot T \cdot R_{f,j}}}{G} \qquad (3)$$

where λ_j is decay constant of radionuclide j, T is time of transport of water (non-retarded species) in geosphere, R_j is retardation coefficient covering also diffusion in matrix of both sorbing and non-sorbing radionuclides and G is factor, which is related to dilution and dispersion of radionuclides in geosphere. Concentration of radionuclides at the interface repository/geosphere can be expressed by the following equation:

$$C_{geo,j}(t) = \frac{\sum_m I_{m,j}(t) \cdot f_{m,j}(t)}{Q_{U/G}} \qquad (4)$$

where $I_{m,j}$ is inventory of radionuclide j in waste form matrices, $Q_{u/g}$ is dilution factor between repository and geosphere, $f_{m,j}(t)$ is fraction of radionuclide j released from repository in one year. This fraction can be expressed by the following equation for radionuclides fixed in matrices

$$f_m(t) = LR_m \cdot F_{uos}(t) \qquad (5)$$

Where LR_m is leaching index of spent fuel matrix or spent fuel structure materials and $F_{uos}(t)$ is fraction of canister failed in time t. For radionuclides, which are not fixed in matrices concentration at the interface repository geosphere is expressed by the following equations:

$$C_{geo,j}(t) = \frac{\sum_m I_{IRF,j}(t) \cdot f_{IRF,j}(t)}{Q_{U/G}} \qquad (6)$$

where $I_{IRF,j}$ is inventory of instant release fraction of radionuclide j in repository and $f_{IRF,j}$ can be expressed by the following equation

$$f_{IRF,i} = F_{\max} \cdot k_{b,j} \qquad (7)$$

where F_{max} is maximal number of canister failed in one year and $k_{b,,j}$ is transfer coefficient of radionuclide j which is indirectly related to residence times of radionuclides in buffer, which in turn are related to apparent diffusion coefficients.

Direct safety functions arranged in hierarchical way and parameters, which will be quantified later and transferred to tentative requirements, are summarized in the following Table 11.

Tab. 11: Hierarchical arrangement of direct safety functions, parameters and components

Level 1	Level 2	Level 3	Level 4	Parameter	Allocated to
Limit releases of radionuclides to the environment by containment and delay				Annual effective dose (250 μSv/yr)	Disposal system (Carbon steel/bentonite/granite concept)
	Limit releases to geosphere by containment and delay			Concentration/flux of radionuclides at interface repository/geospher	Repository (Engineered barrier system
		Limit leaching of radionuclides from spent fuel assemblies		Inventory and leaching rates (I_m I_{IRF}, LR_m)	All waste forms (matrix, structure materials, gap)
			Limit leaching of radionuclides from spent fuel matrix	Inventory in matrix and leaching index	Spent fuel matrix
			Limit leaching from cladding	Inventory in cladding and leaching index	Cladding
			Limit leaching from structure materials	Inventory in structure materials and leaching index	Structure materials
			Limit leaching form gap and grain boundaries	Inventory concentration	Gap and grain boundary
		Isolate radionuclides as long as possible		Canister failure rates (F_{UOS}, F_{max})	Canisters
			Protect carbon steel against contact with water	Failure rate of Ni alloy layer	Ni alloy
			Protect stainless steel against contact with water	Failure rate of Carbon steel layer	Carbon steel
			Protect spent fuel assemblies against contact with water	Failure rate of stainless steel inner	Stainless steel inner canister
			Protect spent fuel rods against contact with water	Failure rate of cladding	Cladding
		Delay release of radionuclides to geosphere		Buffer transfer coefficients ($k_{b,j}$)	Bentonite Buffer
			Provide low solubility in disposal units and interface repository/geosphere	Solubility	Bentonite
			Provide transport by diffusion only	Diffusion coefficients	Bentonite
			Provide high values of sorption	Sorption coefficients	Bentonite
			Provide thickness sufficient for delay by diffusion	Thickness	Bentonite
	Limit transport of radionuclides to biosphere by delay,			Concentration/flux of radionuclides at interface	Granite host rock
		Delay transport of radionuclides to hydraulicaly active zone (HAV)		Concentration and/or flux of radionuclides to HAV	Undisturbed rock zone (UHR)
			Provide long transport of unopsrbed contaminants to hydraulically active zone	Transport time	UHR
			Provide high values of retardation	Retardation coefficients	UHR
			Provide low values of solubilities	Solubility, concentration of colloids	UHR
		Delay transport of radionuclides to the environment		Concentration and/or flux of radionuclides to biosphere	Hydraulically active zone (HAZ)
			Provide long time transport to biosphere	Transport time	HAZ
			Provide high retardation	Retardation coefficients	HAZ
			Provide low values of solubilities	Solubility	HAZ
			Provide dilution and dispersion	Dilution and dispersion coefficient	HAZ

The supporting safety functions for disposal system derived in this work are summarized in the following Table 12.

Tab. 12: Summarization of selected supporting safety functions for concept carbon steel/ bentonite/ granite

Level 1	Level 2	Level 3	Level 4	Parameter	Allocated to
Provide stability against the effect of the environment and human errors (stability/isolation				Set of parameters	Disposal systém
	Provide favourable, stable conditions for EBS			Set of parameters	Host rock
		Provide favourable mechanical conditions		Set of parameters (e.g. maximal shear stress, isostatic load, etc.)	Host rock
			Limit isostatic pressure on disposal units	Isostatic pressure	Host rock
			Limit shear load on disposal units	Shear load	Host rock
		Provide favourable chemical conditions		Set of parameters	Host rock
			Provide low redox conditions	Eh	Host rock
			Provide limited concentration of detrimental agents avoid corrosion rate of canisters, waste form and bentonite	Concentration of detrimental agents	Host rock
			Provide ionic strength to avoid bentonite corrosion	Ionic strength	Host rock
			Provide limited alkalinity to avoid dissolution of buffer	Limit of alkalinity	Host rock
			Provide limited salinity to avoid detrimental effect on buffer swelling	Limit of salinity	Host rock
		Provide transfer of heat from repository to avoid degradation of barriers		Heat transfer properties	Host rock
		Provide favourable hydrogeological conditions for low EBS degradation		Set of parameters	Host rock
		Provide transfer of gases from repository		Set of parameters	Host rock
		Provide favourable microbial conditions		Set of parameters	Host rock
		Provide conditions protecting against criticality		Set of parameters	Host rock
	Provide favourable, stable conditions for host rock and repository itself			Set of parametres	Repository (Engineered Barrier System)
		Prevent mechanical failure		Set of parameters	Repository (Engineered Barrier System)
		Provide low flux of water through repository		Set of parameters	Repository (Engineered Barrier System)
		Provide favourable chemical conditions for disposal system safety functions		Set of parameters	Repository (Engineered Barrier System)
		Provide favourable thermal properties for heat transfer from a repository		Set of parameters	Repository (Engineered Barrier System)
		Provide favourable microbiological properties for repository safety functions and host rock itself		Set of parameters	Repository (Engineered Barrier System)
		Limit amount of gases originating in a repository and enable transfer of gases through repository		Set of parameters	Repository (Engineered Barrier System)
		Provide conditions unfavourable for critical reactions		Set of parameters	Repository (Engineered Barrier System)

The quantitative requirements are similar to safety function criteria derived in Swedish safety case [11) (Fig. 4). In Czech disposal programme, they will be established in future projects after agreement with RAWRA and on the basis of expert elicitation approach.

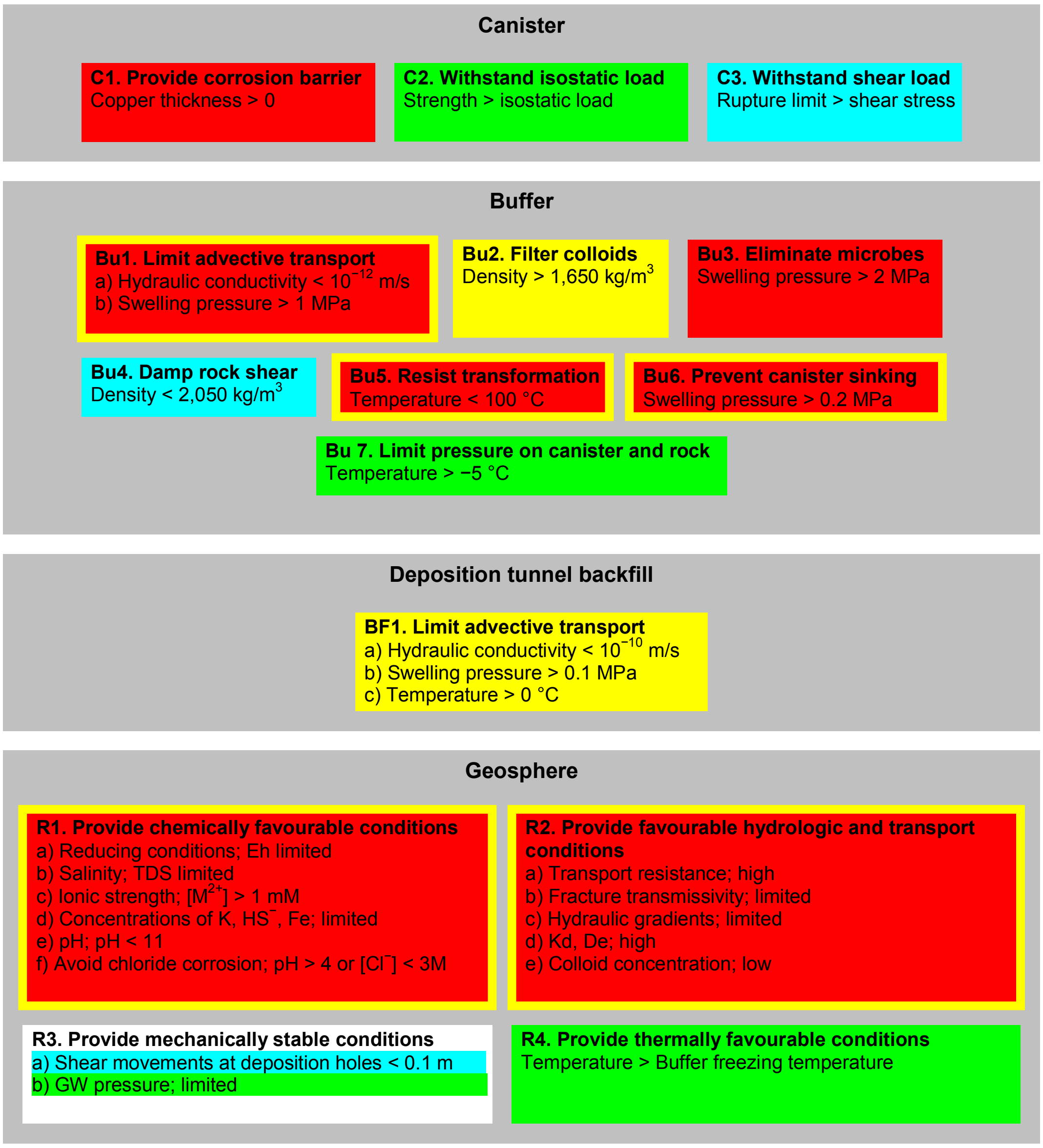

Fig. 4: Safety functions, safety function indicators and safety function indicator criteria for Swedish SR-CAN report [11]

When quantitative criteria cannot be given, terms like “high”, “low” and “limited” are used in Swedish approach to indicate favourable values of the safety function indicators. The colour coding shows how the functions contribute to the canister safety functions C1 (red), C2 (green), C3 (blue) or to retardation (yellow). Many functions contribute to both C1 and retardation (red box with yellow board).

6 Derivation of main categories of scenarios leading to the failure of disposal system

A consensus was reached in WP 1 among the participating organizations 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. It was shown, however, that the definition and use of terminologies is dependent on specific national frameworks and safety case methodologies. 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 organizations consider a base case, which describes a starting point for scenario development. The participants called this overall case "central scenario". 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.

The procedure of scenario development itself has also some components which are widely used by various organizations. 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

The approach used in this work does not differ very much from this approach, the most important difference being that first the collection and screening of FEPs and other knowledge is used for identification of safety functions. Basic scenarios are then derived from analysis of the effect of surrounding systems on safety functions of disposal system, subsystems and components.

Safety functions are questioned in relation to initiating event and processes both from external and internal systems.

At the first level there is only a disposal system with safety function *F.1 Limit releases of radionuclides to the environment by containment and delay,* which interact with the surrounding systems: the environment and development and construction system. The normal evolution scenario is defined for expected normal evolution of disposal system.

The following basic assumptions have been accepted for Czech disposal system for safety functions FS1 – FS3 for normal evolution.

1) The environment conditions in future years will correspond to natural changes in a site corresponding to changes in last 100 000 years (earthquakes intensity, climate change, etc) and no human intrusion will occur under normal repository evolution. It is also specified by site exclusion criteria given above (FS2).

2) No errors of disposal system will occur due to the development, design, construction or component manufacture error more than "designed" (sealing system, canister premature failures, etc) will occur under development, construction and operational phases of the disposal system. It will be ensured by careful QA system (FS3)

3) Function properties of repository itself will be in some design range for assessment period (FS1)

These three basic assumptions form a baseline for normal/central evolution scenario. The failure of the assumptions will lead to other scenarios. In the case, where it is not possible to quantify probability or to determine the time of the event, these scenarios can be evaluated as stylised scenarios.

The basic 6 categories of scenarios derived from analyses of External FEPs from OECD/NEA database [10] are given in Table 13.

The external factors taken from the international or national databases will be then judged according to relevancy to:

- Repository design (geology, selected barriers)
- Waste
- Geographic setting
- Probability of occurrence in given time of assessment (if it is not possible to quantify the probability and it is not possible to say that it is zero, then scenarios can be evaluated as stylised scenario)
- Consequence of the external effects (e.g. negligible, low, medium, high, very high with probable time of occurrence if it is possible)

The stylized scenarios for human intrusion are dealt in other part of the project.

Tab. 13: Basic categories of scenarios derived from analyses of external FEPs of OECD-NEA database [10]

Scenario category number	Title	Description
1	Normal evolution scenario	Climatic changes and natural and human errors and activities has no significant effect on repository under normal evolution scenario of repository, it is not supposed any deviation from designed properties of system components
2	Geological processes and events scenarios	A geological event or process will affect unfavourably disposal system (e.g. preferential path, premature failure of engineered barrier) (OECD/NEA - FEPS code 1.2)
3	Climate change scenarios	Climatic changes will lead to significant change of conditions, leading to the conditions in a repository (OECD/NEA - FEPS code 1.3)
4	Repository issues scenarios	Human error relating to development and construction activities will lead to a significant change of conditions and premature failure of components, preferential paths, criticality etc., under these scenarios can be also considered insufficient knowledge of processes occurring in a repository or site. For example badly characterised site can lead to some fast preferential paths leading from a repository to the environment. (OECD/NEA - FEPS code 1.1)
5	Future human actions scenarios	Human intrusion can lead to significant change of conditions, failure of barriers or direct contamination of people (OECD/NEA - FEPS code 1.4)
6	Disposal system evolution scenarios	Scenarios initiated by processes and/or events occurring in a disposal domain (e.g. criticality)

7 Scenario specification using safety function analyses

Safety functions of a disposal system determine what the system does and supporting safety functions conditions under which these safety functions works. It can be expected that if direct and supporting safety functions of all subsystems and components will work then also the whole disposal system will work. Bottom up analyses will start from the lowest level corresponding to the lowest level of system decomposition achieved for normal evolution scenario. Normal evolution scenario is composed of a number of normal, expected evolution of subsystems and components of disposal system. It requires evaluation of possible states of their evolution on fulfilling their safety function and possibly identification of further direct or supporting safety functions.

Altered scenarios will start from the analysis of the effect of initiating event on safety functions of subsystems and components. For example “Earthquake scenario” will require an analysis of possible earthquake with magnitude exceeding the magnitude of “designed earthquake” for normal evolution scenario on safety functions of subsystems and their components in bottom-up direction.

If the probability of initiating events and their frequency can be estimated, it can be calculated as altered scenario, but if the probability is very low and cannot be quantified then it can be treated as stylised scenario or possibly what-if scenario calculated for demonstration of safety even if it is often physically impossible.

8 Conclusion

In this work one of the systems engineered approach used in managing complex projects was applied to derive safety functions and scenarios for evaluation of safety of concept of disposal of spent fuel assemblies in carbon steel canisters surrounded by bentonite buffer in granite host rock. The main principle of the approach is to define first safety functions of a disposal system, subsystems and its components in top-down, hierarchical arrangement, formulate requirements and then to analyze these safety functions of systems, subsystems and components in respect of all possible interactions with surrounding systems, subsystems and components in bottom-up direction.

The safety functions were divided into two categories. Direct safety functions strictly related to the main objective of the disposal system to limit the releases to biosphere by containment and delay so that the exposure of people is lower than limits of regulatory body belong to the first category and supporting safety functions which cover all the assumptions under which the direct safety functions must work. Both safety functions categories are arranged hierarchically so that upper level functions provide scope for next level.

The requirements for direct safety functions are linked closely to parameters used in safety calculations. The requirements have, however, only tentative character. It means that they can be changed with increasing knowledge of the system.

Six basic scenarios which include normal evolution scenario and scenarios initiated by an event or process from interaction of disposal system with external systems (environment and development and construction activities) were defined. These scenarios will be evaluated by systematic analysis of the effect of initiating event or processes on safety functions of concrete subsystems and components.

One of the most important advantages of this approach is that all safety functions and requirements are linked hierarchically to the main objective of the system so that there is less probable that some important aspect will be neglected.

9 Literature

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Part 2: Stylised Scenarios

Part 2.1: Regulatory Perspective of Stylised Human Intrusion Scenarios

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Executive Summary

This document reports on activities performed within Task 12 of PAMINA WP3.1 by Galson Sciences Limited (**GSL**). WP3.1 comprises two complementary parts: the identification of scenarios based on safety functions; and the development of stylised scenarios. Task 12, this report, explores the regulatory requirements for stylised human intrusion scenarios.

The nature and probability of future human actions that could disturb a disposal facility are highly uncertain, so assessments are based on the assumption that human behaviour and technological capability in the future will be largely the same as today – in order to avoid undue conjecture. While direct human intrusion into a geological disposal facility is unlikely to occur, direct intrusion could result in significant exposure to the intruders, migration of radionuclides into the accessible environment, and increased exposures that are indirectly related to the intrusion. Similar concerns apply to near-surface disposal facilities. It is therefore considered necessary to assess the potential consequences of human intrusion events.

The International Commission on Radiological Protection (ICRP) set out its recommendations for the regulatory consideration of human intrusion in 1998 (ICRP Publication 81). The ICRP considers it inappropriate to include the probability of future human actions in a quantitative performance assessment for comparison with dose or risk constraints. Instead, the consequences of one or more stylised scenarios should be considered to evaluate the resilience of the disposal system design to such events. ICRP Publication 103, published in 2007, contains updated general radiological protection principles, but regards the advice in ICRP Publication 81 as still valid, including the view that efforts to reduce the probability and/or consequences of human intrusion events are the best means of achieving radiological protection.

The development of similar human intrusion scenarios in several European countries and for the US Waste Isolation Pilot Plant (WIPP) Project provides confidence that the scenarios are appropriate for many sites. Aspects of a site that need to be taken into consideration when developing human intrusion scenarios include the economic potential of the host rock and likely future human activities in the area. These factors will affect the probability of drilling, mining, water abstraction and construction activities. As the WIPP is located in a resource-rich area, the regulators specified that mining and drilling scenarios should be considered in the performance assessment.

ICRP policy on human intrusion has helped shape the UK environment agencies' approach in providing new (2009) regulatory guidance on the treatment of future human actions in safety assessments of disposal facilities, and this presents a largely consistent picture with current European regulatory views on this issue. There are some differences in approach to the regulatory treatment of future human actions between countries but, within Europe, there is good consensus on the following points:

- Actions should be taken during siting and design of disposal facilities to reduce the probability and/or consequences of potentially disruptive future human actions.

- Assessments of future human actions should be based on present-day conditions in the region of the disposal site and at similar sites. Assessments should assume present-day social structures and technological capabilities to avoid undue conjecture.

- Assessments of disposal system performance without disruptive future human actions should include the effects of any recent and ongoing human activities that might affect the performance of the disposal system. The potential effects of disruptive future human actions should be assessed using separate scenarios, which may be proposed by the developers/operators or the regulator. Scenario probability should only be considered qualitatively.
- Assessments should exclude consideration of deliberate intrusion, and should consider only inadvertent intrusion scenarios.

Regulatory requirements for the WIPP differ from the regulatory consensus in Europe in that the regulator defined the procedures and assumptions necessary for the developer/operator to quantitatively determine the probability of occurrence of future human intrusion scenarios.

1 Introduction

1.1 Background and Aims

PAMINA (**P**erformance **A**ssessment **M**ethodologies **IN** **A**pplication to Guide the Development of the Safety Case) is an Integrated Project funded by the Sixth Framework Programme of the European Commission. The work is organised in four Research and Technology Development Components (RTDCs) and one additional component dealing with knowledge management and dissemination of knowledge. Research and Technology Development Component 3 (RTDC-3) aims to develop methodologies and tools for integrated performance assessment (PA) for various geological disposal concepts. It consists of four Work Packages (WPs): the development of PA scenarios (WP3.1), the PA approach to gas migration processes (WP3.2), the PA approach to radionuclide source term modelling (WP3.3), and safety and performance indicators (WP3.4).

WP3.1 comprises two complementary parts: the identification of scenarios based on safety functions; and the development of stylised scenarios. It has been divided into 21 tasks, each of which will result in a Milestone report. Task 12, this Milestone report, explores the regulatory requirements for stylised human intrusion scenarios.

Future human intrusion can be defined as:

- Deliberate or inadvertent intrusion directly into a disposal facility.
- Deliberate or inadvertent intrusion that damages natural or engineered barriers or degrades their functions.

There is a general consensus that human intrusion should be considered as part of the safety case for the disposal of solid radioactive waste in both geological and near-surface disposal facilities (e.g. IAEA, 2000).

A stylised approach is likely to be used for human intrusion because of the uncertainty inherent in trying to predict what people might do in the future. Where few or no relevant data are available, a stylised approach can use arbitrary assumptions that '*are plausible and internally consistent, but err on the side of conservatism*' (Environment Agency and NIEA, 2009).

This report aims to provide a regulatory perspective on the treatment of human intrusion in the safety case.

1.2 Methodology and Report Structure

There are differences between national regulatory requirements with respect to the assessment of human intrusion in performance assessment, some of which are the result of revisions to the International Commission on Radiological Protection (ICRP) recommendations for radiological protection of members of the public following closure of a radioactive waste disposal facility.

Section 2 summarises the ICRP principles of protection and the guidance on human intrusion.

Section 3 discusses various regulatory approaches to the treatment of human intrusion. This is done by presentation of a detailed example of the incorporation of the ICRP recommendations on human intrusion into regulatory guidance by the UK environment agencies (Environment Agency and NIEA 2009; Environment Agency *et al*., 2009). The findings of the European Pilot Study in relation to human intrusion are then discussed (Vigfusson *et al*., 2007). The European Pilot Study is important because it represents the combined views of regulatory bodies and technical support organisations from seven European countries (Belgium, France, Germany, Spain, Sweden, Switzerland and the UK). Finally, we discuss an alternative approach to regulating future human intrusion used for the US Waste Isolation Pilot Plant (WIPP) Project.

Section 4 identifies and discusses good practice for the consideration of human intrusion in regulation and in safety cases.

Section 5 lists the references used in the report.

2 ICRP Guidance

2.1 Introduction

The ICRP provides recommendations and guidance on protection against the risks associated with ionising radiation. The most recent set of '*Recommendations for a System of Radiological Protection*', including the ICRP principles of protection, was published in 2007 (ICRP Publication 103; ICRP, 2007). The ICRP considers that the principles are fundamental to the system of protection from exposure. In Section 2.2, the ICRP principles of protection are briefly summarised.

The ICRP published a set of recommendations specifically relevant to the regulatory consideration of potential future human intrusion at radioactive waste disposal sites in ICRP Publication 81 (ICRP, 1998). These recommendations apply to the radiological protection of members of the public from the near-surface or geological disposal of long-lived solid radioactive waste in new disposal facilities. Section 2.3 describes the application of the radiological protection criteria to disposal facilities.

2.2 ICRP Principles of Protection

The ICRP considers that the following principles of protection from exposure for members of the public should be applied across a wide range of planned, emergency and existing exposure situations (ICRP, 1998, 2007):

- **Justification of a practice** – any decision that alters the radiation exposure situation should do more good than harm. The waste management and disposal operations should be included in the assessment of the justification of the practice generating the waste.

- **Optimisation of protection** – the likelihood of incurring exposures, the number of people exposed, and the magnitude of their individual doses should all be kept as low as reasonably achievable, taking into account economic and societal factors.

- **Application of dose limits** – the total dose to any individual from regulated sources in planned exposure situations (other than medical exposure of patients) should not exceed the appropriate limits recommended by the ICRP Control of public exposure through a process of constrained optimisation will '*obviate the direct use of the public exposure dose limits in the control of radioactive waste disposal*' (ICRP, 1998).

2.3 Application of ICRP Recommendations to the Disposal of Long-lived Solid Radioactive Waste

The recommendations in ICRP Publication 81 (ICRP, 1998) apply to the radiological protection of members of the public from the near-surface or geological disposal of long-lived solid radioactive waste in new disposal facilities. The ICRP distinguishes between normal and potential exposures: '*normal exposures*' are virtually certain to occur and have a magnitude which is predictable, albeit with some uncertainty; '*potential exposures*' arise from situations where there is no certainty regarding the likelihood of occurrence of an exposure or its magnitude. The ICRP definition of '*normal*' exposures applies to exposures involving the deliberate introduction and operation of sources, such as at nuclear power stations or medical facilities. Following closure of a radioactive waste disposal facility, there will not be any deliberate releases of radionuclides and therefore all exposures will be '*potential*', whether resulting from natural processes or human intrusion.

ICRP Publication 77 highlighted the problems of estimating doses over extended periods: 'both the individual doses and the size of the exposed population become increasingly uncertain as time increases. Furthermore, the current judgements about the relationship between dose and detriment may not be valid for future populations.' (ICRP, 1997). The principal means of protecting the public from radiation exposures in the long periods involved is through a process of constrained optimisation, taking account of the ICRP's recommended upper value for the individual dose constraint of 0.3 mSv/a or its risk equivalent (ICRP, 1998). However, the ICRP recommends that radiological protection criteria are applied differently to potential exposures that arise as a result of natural processes (Section 2.3.1) and as a result of future human intrusion (Section 2.3.2).

2.3.1 Radiological Protection of Members of the Public from Potential Exposures as a Result of Natural Processes

The ICRP (1998) considers that, in situations where potential exposures result from natural processes, assessed doses or risks should be compared with a dose constraint of no more than about 0.3 mSv/a or its risk equivalent of around 10^{-5} per year. These natural processes may include the gradual degradation of waste packages and the consequent release and transport of radionuclides in groundwater, or less likely but more disruptive processes such as seismic events and glaciation.

The approaches to show whether constraints are satisfied can be either a calculation of risk by combining doses and probabilities or, for each exposure situation, a presentation of the dose and its corresponding probability of occurrence separately. The same degree of protection can be achieved by using either a dose constraint or its risk equivalent, supplemented by a consideration of the probability that the doses would be incurred. However, more information may be obtained for decision-making purposes by adopting the latter approach.

2.3.2 Radiological Protection of Members of the Public from Potential Exposures as a Result of Future Human Intrusion

The ICRP (1998) position on future human intrusion is summarised in this section. Human intrusion into a disposal facility containing concentrated radioactive waste may result in waste material being brought to the surface and directly exposing nearby populations to significant radiation doses. Radionuclides released following human intrusion could also migrate into the accessible environment, resulting in exposures that are indirectly related to the intrusion. With regard to inadvertent human intrusion, such as drilling into a geological disposal facility after closure or disturbance of a near-surface disposal facility during future construction activities at the site, the ICRP considers that the consequences of one or more plausible stylised scenarios should be considered to evaluate the resilience of the disposal system design to such events.

Compared to natural processes, little scientific basis exists for predicting the nature or probability of future human actions. Therefore, the ICRP considers that it is not appropriate to calculate the probabilities of such events in a quantitative performance assessment for comparison with risk constraints, and do not recommend that the individual maximum dose constraint of 0.3 mSv/a for members of the public is used to evaluate the significance of human intrusion.

The ICRP considers that a performance measure appropriate to evaluating the significance of future human intrusion can be based on that which would apply to controlling exposures under present-day circumstances. Accordingly, the ICRP recommends that an annual dose of around 10 mSv/a may be used as a generic reference level below which intervention to control exposures is not likely to be justifiable. The ICRP also considers that in circumstances where human intrusion may result in sufficiently high doses (an annual dose of around 100 mSv/a may be used as a generic reference level), '*intervention on current criteria would almost always be justified*'.

Protection from exposures associated with future human intrusion may be best achieved, according to the ICRP, by efforts to reduce the probability and/or consequences of such events. These efforts may include taking reasonable measures to warn society of the existence of the disposal facility, employing active or passive institutional controls, locating the facility at depth, and incorporating features into the design of the disposal facility to make intrusion more difficult. In addition, for the near-surface disposal of radioactive waste, the inventory of the facility may be controlled so that it contains only low level or short-lived intermediate level waste.

In summary, the ICRP recommendations state that it is inappropriate to include the probability of future human actions in a quantitative performance assessment for comparison with dose or risk constraints. Instead, the consequences of one or more stylised scenarios should be considered to evaluate the resilience of the disposal system design to such events.

3 Recent Examples of Regulatory Application

The UK environment agencies' Guidance on Requirements for Authorisation documents (the GRAs) for geological disposal facilities (Environment Agency and NIEA, 2009) and for near-surface disposal facilities (Environment Agency *et al.*, 2009) are the most recent examples of regulatory requirements for radioactive waste disposal developed using the latest ICRP guidance. The GRAs apply different criteria to the assessment of natural processes (Section 3.1), compared to the treatment of future human intrusion (Sections 3.2 and 3.3). A European Pilot Study of regulatory approaches also recommended the use of a stylised approach to the treatment of future human intrusion (Section 3.4). Finally, we summarise the approach to regulation at the WIPP (Section 3.5). This is the only geological disposal facility currently in operation and is unique in its approach to the consideration of future human actions in performance assessment.

3.1 Application of a Risk Guidance Level to the Assessment of Natural Processes (UK)

The environment agencies in the UK consider that the effective dose from a disposal facility to a representative member of the critical group should be As Low As Reasonably Achievable (ALARA) taking into account economic and societal factors. Requirement 5 of the GRA states that during the period of authorisation of a disposal facility (prior to the withdrawal of control), the effective dose to a representative member of the critical group should not exceed a source-related dose constraint (0.3 mSv/a) and a site-related dose constraint (0.5 mSv/a).

Following the withdrawal of control from a disposal facility, the UK environment agencies consider that the assessed radiological risk from the disposal facility to a person representative of those at greatest risk should be consistent with a 'risk guidance level' of 10^{-6} per year (Requirement R6). The risk guidance level is defined as '*a level of radiological risk from a disposal facility which provides a numerical standard for assessing the environmental safety of the facility*' and is consistent with advice given by the UK Health and Safety Executive (HSE, 2001) and the Health Protection Agency (HPA, 2009). The developer/operator of the disposal facility is expected to produce risk assessments that demonstrate consistency with the risk guidance level; however, this approach does not apply to human intrusion, which is considered separately (Figure 3.1).

Different approaches are taken to the treatment of human intrusion by the GRAs for geological and near-surface disposal facilities; these are described in Sections 3.2 and 3.3 respectively.

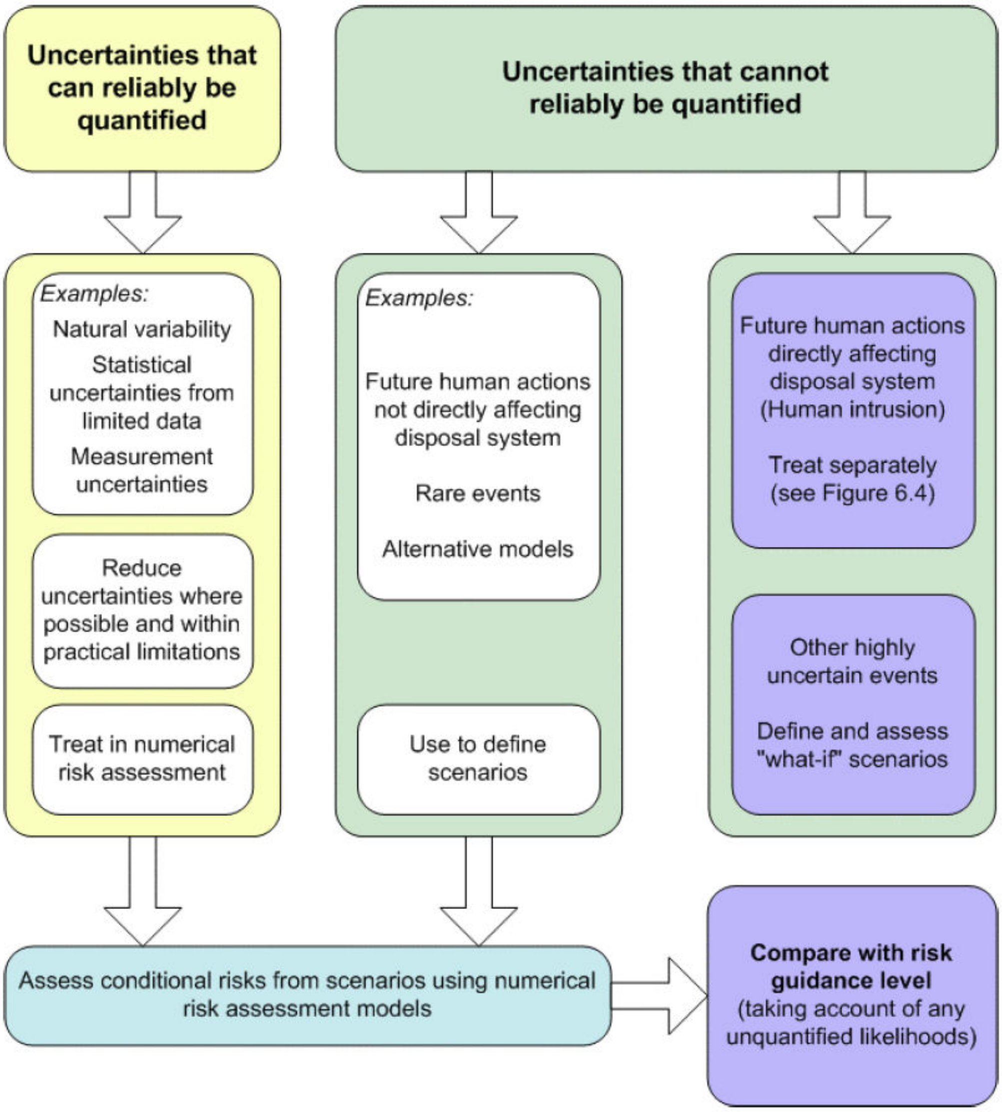

Fig. 3.1: Approach in the UK to the treatment of uncertainties taken by the GRAs for geological and near-surface disposal facilities (Environment Agency and NIEA, 2009; Environment Agency *et al*., 2009)

3.2 Regulatory Consideration of Future Human Actions for Geological Disposal (UK)

The Environment Agency and NIEA (2009) consider that potential future human actions affecting a geological disposal facility are associated with uncertainties that cannot reasonably be quantified. These unquantifiable uncertainties will need to be taken into account in developing a safety case for the geological disposal of radioactive wastes, but should be given separate consideration to quantifiable uncertainties, such as those related to some natural processes (Figure 3.1).

Accordingly, Requirement R7 in the GRA for geological disposal states that uncertainties associated with human intrusion after the period of authorisation should be treated separately, and specifically that:

‘The developer/operator of a geological disposal facility should assume that human intrusion after the period of authorisation is highly unlikely to occur. The developer/operator should consider and implement any practical measures that might reduce this likelihood still further. The developer/operator should also assess the potential consequences of human intrusion after the period of authorisation.’

Note that the GRA for geological disposal does not include any reference to the generic reference levels given for annual doses in the ICRP recommendations (Section 2.3.2). For a geological disposal facility, human intrusion may be regarded as highly unlikely (but not impossible), because the depth of disposal is expected to put the facility beyond the reach of most types of human activity. There can be no guarantee of protection for anyone coming into direct contact with high-level waste; such individuals could, in principle, receive a radiation dose up to and including a fatal dose.

Whereas for a near-surface disposal facility, it is possible to limit the inventory of the facility, this is not possible for a geological disposal facility, in which it is the activity of the waste form rather than the overall inventory that is the controlling factor. The Environment Agency and NIEA (2009) therefore do not apply the ICRP reference levels for annual doses. Rather, an assessment of radiation doses to individuals representative of those undertaking the intrusion and those who might subsequently be exposed to additional dose as a result of an intrusion event is required. The results should be used in optimisation studies that allow the developer/operator to ‘*reduce the radiological impacts resulting from human intrusion, subject to balancing all the other considerations relevant to optimisation*’.

The environment agencies have classified human intrusion as:

1. Intrusion with full knowledge of the existence, location, nature and contents of the disposal facility.
2. Intrusion without prior knowledge of the disposal facility, e.g. drilling a well or exploratory drilling for mineral resources based on an assessment of the local geology.
3. Intrusion with knowledge of the existence of underground workings but without understanding what they contain, e.g. an archaeological investigation carried out without knowledge or understanding of radioactivity.

The developer/operator is expected to consider the second and third classes of scenario.

The Environment Agency and NIEA (2009) consider that human intrusion may be ‘directly into a disposal facility’ or ‘human actions that damage barriers or degrade their functions, such as partial excavation of a previously closed and sealed access tunnel or shaft’. Both natural and engineered barriers may be affected by future human actions; the extent to which human actions affecting the surrounding host rock should be considered against the risk guidance level (10^{-6} per year) or using a stylised approach would be decided during discussions between the environment agencies and the developer/operator.

The Environment Agency and NIEA (2009) consider that the timing, type and extent of human intrusion into a geological disposal facility are so uncertain that they should be explored through one or more ‘what-if’ scenarios, separate from the scenarios representing the undisturbed evolution of the disposal system. Scenarios should be based on human actions that use current or historical techniques and technologies, similar to those applied in similar geological and geographical settings. These scenarios can be used to assess the consequences of human intrusion into a geological disposal facility and should include all human actions associated with any material removed from the facility, including considering what is then done with this material.

The Environment Agency and NIEA (2009) would expect the developer/operator of a geological disposal facility to consider and implement measures that might reduce the likelihood and/or consequences of human intrusion, where these measures are practical and are likely to be beneficial. It is particularly important that measures intended to reduce the likelihood of human intrusion do not compromise the environmental safety performance of the disposal system if human intrusion does not occur. The number of people involved in actions associated with the intrusion, and the number of people exposed as a result of occupying the site or surrounding area following the intrusion should also be assessed, and optimisation principles used to reduce the radiological impact to humans and non-human biota resulting from human intrusion.

Assessments of the radiation dose to individuals undertaking the intrusion and those who may occupy the site or neighbourhood following intrusion should be presented by the developer/operator. These assessments should take into account all radionuclides present, and all decay products that contribute significantly to dose, and heterogeneity in the waste. They should also investigate the consequences of intrusion on the long-term behaviour of the disposal system and on the wider geographical area.

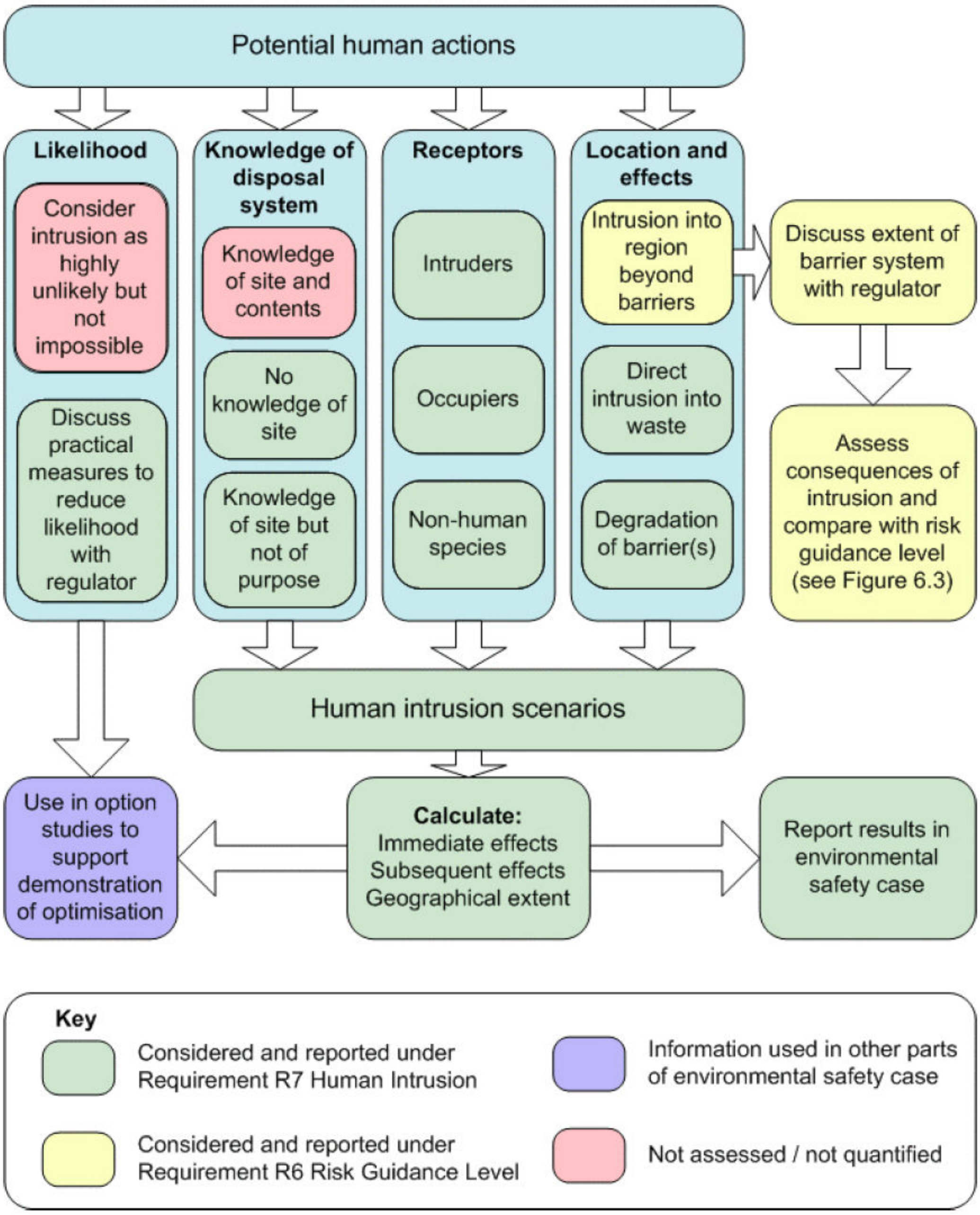

Fig. 3.2: Approach in the UK to the treatment of potential future human actions after the period of authorisation for a geological disposal facility (Environment Agency and NIEA, 2009).

The regulatory guidance on the treatment of future human actions for a geological disposal facility is summarised in Figure 3.2.

3.3 Regulatory Consideration of Future Human Actions for Near-surface Disposal (UK)

The UK environment agencies consider potential future human actions affecting near-surface and geological disposal facilities in a different way. Their position with respect to near-surface disposal facilities is summarised in this section. Near-surface disposal facilities are defined as those located at the surface of the ground, or at depths down to several tens of metres below the surface (Environment

Agency *et al*., 2009). Requirement R7 in the GRA for near-surface disposal of solid radioactive waste states that:

'The developer/operator of a near-surface disposal facility should assess the potential consequences of human intrusion into the facility after the period of authorisation on the basis that it is likely to occur. The developer/operator should, however, consider and implement any practical measures that might reduce the chance of its happening. The assessed effective dose to any person during and after the assumed intrusion should not exceed a dose guidance level in the range of around 3 mSv per year to around 20 mSv per year. Values towards the lower end of this range are applicable to assessed exposures continuing over a period of years (prolonged exposures) while values towards the upper end of the range are applicable to assessed exposures that are only short-term (transitory exposures).'

The UK environment agencies consider that the developer/operator will not 'be able to substantiate that human intrusion into a near-surface disposal facility is unlikely to occur after the period of authorisation'. Wastes in a near-surface disposal facility are potentially vulnerable to disturbance by ordinary human actions. Human intrusion into a near-surface disposal facility is assessed against a 'dose guidance level', specified above in Requirement R7, rather than a risk guidance level, because the likelihood of human intrusion cannot reliably be assessed in terms of a numerical value of probability. It should be noted that the dose guidance level range of around 3 mSv/a to around 20 mSv/a adopted in the UK implies that interpretation of ICRP recommendations on the treatment of human intrusion for near-surface disposal facilities may differ between countries.

Human intrusion for a near-surface disposal facility is classified in the same way as for a geological disposal facility, with the appropriate method of assessment of intrusion into natural barriers that provide environmental safety functions decided in discussions between the environment agencies and the developer/operator.

The timing, type and extent of human intrusion into a near-surface disposal facility are so uncertain that they should be explored through one or more 'what-if' scenarios, separate from the scenarios representing the undisturbed evolution of the disposal system. The recommended methodology is comparable to that used for geological disposal facilities – see Figure 3.3.

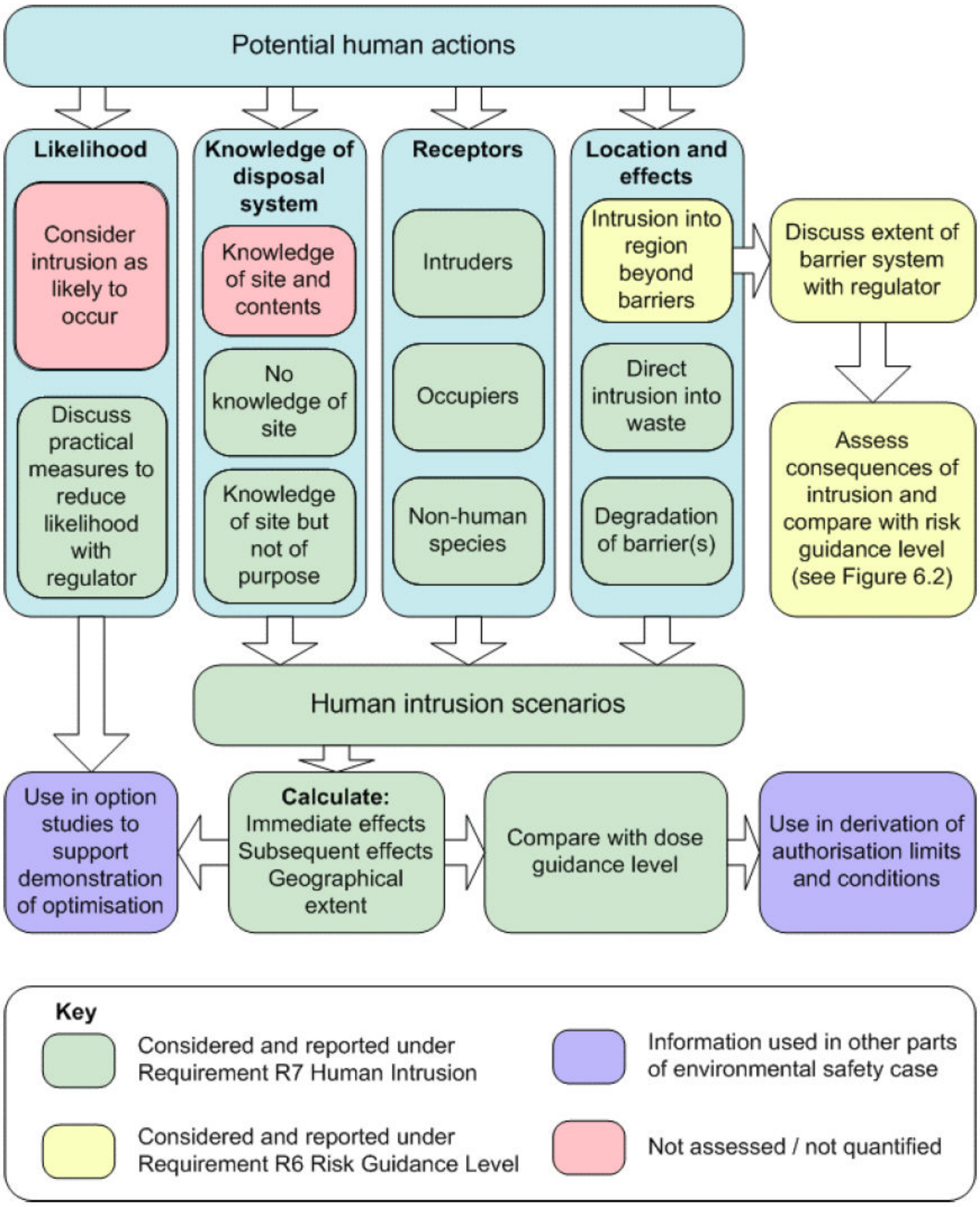

Fig. 3.3: Approach in the UK to the treatment of potential future human actions after the period of authorisation for a near-surface disposal facility (Environment Agency et al., 2009).

The environment agencies would expect the developer/operator of a near-surface disposal facility to consider and implement practical measures that might reduce the likelihood of human intrusion, where these measures are practical and are likely to be beneficial. In common with geological disposal, it is particularly important that measures intended to reduce the likelihood of human intrusion do not compromise the environmental safety performance of the disposal system if human intrusion does not occur. For many wastes considered suitable for near-surface disposal, human intrusion after the period of authorisation is likely to result in doses below the dose guidance level. The environment agencies state that in such cases, they are unlikely to '*insist on any but the simplest measures for reducing the likelihood of intrusion*'.

The number of people involved in actions associated with the intrusion, and the number of people who might be exposed as a result of occupying the site or surrounding area following the intrusion should also be assessed and optimisation principles used to reduce the radiological impact to humans resulting from human intrusion. Assessments of the radiation dose to individuals undertaking the intrusion and those who may occupy the site or neighbourhood following intrusion should be presented by the developer/operator to the relevant environment agency. Assessments of the radiation doses received by non-human biota as a result of human intrusion should demonstrate that these are not at a level likely to cause significant harm to populations of biota. The developer/operator should also demonstrate that intrusion by non-human species, including plant species (e.g. tree roots), is not a significant issue.

The environment agencies consider that assessments should take into account all radionuclides present, all decay products that contribute significantly to dose, and heterogeneity in the waste. The assessments should also investigate the consequences of intrusion on the long-term behaviour of the disposal system and on the wider geographical area. For wastes with a significant inventory of long-lived radionuclides, potential doses around the dose guidance levels are possible for scenarios in which significant amounts of waste are disturbed. The results of these scenarios are likely to contribute to the derivation of facility-specific authorisation limits, such as inventory limits and allowable activity concentrations for specific radionuclides.

3.4 Wider Regulatory Attitudes in Europe

A European Pilot Study of regulatory approaches to the review of safety cases for the geological disposal of radioactive waste (Vigfusson *et al*., 2007) identified broadly similar regulatory attitudes as those in the UK to the achievement and management of safety in the context of the management of uncertainties. The European Pilot Study is important because it represents the combined views of regulatory bodies and technical support organisations from seven European countries (Belgium, France, Germany, Spain, Sweden, Switzerland and the UK).

Vigfusson *et al*. (2007) considered that future human behaviour is unpredictable and, therefore, the concept of minimising uncertainties is meaningless in this context. However, steps can be taken in the choice of disposal facility location or design to limit the potential for, and consequences of, human actions affecting the disposal facility. A range of future human actions has the potential to breach or significantly degrade the natural and/or engineered barriers of the geological disposal facility. It is not considered necessary to assess the risks of deliberate human intrusion, since the responsibility for these actions lies with the people taking them, who would pay due regard to safety, economic and environmental implications (NEA, 1995).

Although there is a very low probability of inadvertent human actions affecting a geological disposal system, in the longer term institutional controls are unreliable and the developer will be expected to assess the consequences of future human actions. Any view on the likelihood of such actions will be highly speculative and therefore may be of little value. For this reason, the study concluded that:

'An exploration of future human actions, including human intrusion, needs to be based on a stylised approach in which one or more scenarios are generated making arbitrary assumptions. Any assessment of the likelihood of such scenarios will also be arbitrary.'

Assumptions for future human actions include the availability of present-day technology for intrusive drilling or mining. This is considered to be a conservative assumption if it is also assumed that the location or purpose of the geological disposal facility is unknown, but the assumption is made to avoid undue conjecture. Although there will be regional differences in the proposed scenarios, they may include:

1. Drilling for water into a contaminant plume from the geological disposal facility.
2. Exploratory drilling into the geological disposal facility or plume with the extraction of cores.
3. The operation of a mine near the geological disposal facility.
4. Direct physical human intrusion into the geological disposal facility.

In France, new guidelines were issued by the regulator (ASN, Nuclear Safety Authority) in 2008. These are detailed in Part 2.2 (Section 2.6) of this report, and include requirements to consider climate change and seismic activity scenarios as part of the evaluation of the consequences of natural climatic changes (ASN, 2008).

The potential consequences of radiation exposure to intruders could be significant, the consequences for other exposed groups of people and the wider environment from radionuclides transported past breached or degraded barriers are likely to be much less severe (Vigfusson *et al*., 2007). It would be difficult to reduce the dose received by direct intrusion by modifying the facility design, and it would therefore not normally be appropriate to compare this dose with a regulatory limit. Minimising the likelihood of intrusion by selecting a site that is not rich in natural resources and locating the facility at depth would be appropriate. Also, understanding the indirect effects of human intrusion (e.g. degraded or breached barriers) may feed into the optimisation of the facility design to make it more robust. However, care needs to be taken to ensure that this aspect of optimisation is not overly dependent on the stylised assumptions made.

3.5 Regulatory Requirements for Human Intrusion at the US Waste Isolation Pilot Plant (WIPP)

The WIPP, located in New Mexico, is operated by the US Department of Energy (DOE) and certified by the US Environmental Protection Agency (EPA). It is a geological disposal facility located at a depth of 655 m in a Permian bedded salt formation, and used for the disposal of transuranic (TRU) defence wastes. The EPA published generic regulations for radioactive waste disposal in 1993 (at 40 CFR Part 191), and WIPP-specific regulations (the WIPP Compliance Criteria) in 1996 at 40 CFR Part 194.

The regulatory criteria for the WIPP are unique in requiring quantitative assessment of the impacts of future human actions, considering the probability of occurrence – and significant regulatory guidance is given on how to calculate both probability and consequence. A key reason for the different approach at the WIPP is the location of the facility in a resource-rich sedimentary basin, with the potential for extractable resources both above (potash) and below (hydrocarbons) the disposal horizon. The regulations in 40 CFR Part 194 specify that performance assessments are required to *'consider natural processes and events, mining, deep drilling and shallow drilling that may affect the disposal system during the regulatory time frame*' (10,000 years after disposal; US EPA, 1996, §194.32). The 40 CFR Part 194 criteria are site-specific and because the WIPP is located in a resource-rich area, the procedures and inherent assumptions for defining the likelihood of occurrence of human intrusion scenarios are prescribed in detail in the regulations.

The DOE divided human activities into three categories: (i) human activities that are currently taking place and those that took place prior to submission of the Compliance Certification Application (CCA); (ii) human activities that might be initiated in the near future; and (iii) human activities that might be initiated after the closure of the disposal facility. The EPA specifies that consideration of future human actions is limited to current drilling and mining practices, so, for example, the DOE were able to eliminate drilling associated with geothermal energy production, hydrocarbon production or archaeological investigation from consideration (Galson *et al*., 2000).

Future human activities included in performance assessment calculations for the WIPP are associated with mining and deep drilling within the controlled area (41 km^2 zone above the WIPP) at a time when institutional controls cannot be assumed to eliminate completely the possibility of such activities. All other future human activities have been eliminated from the performance assessment by the regulator, or screened out by the developer/operator based on low consequence or low probability. For example, the regulations specify that intentional intrusion into the WIPP should be excluded from assessment calculations; and the effects of shallow drilling within the controlled area were eliminated by the developer/operator on the basis of low consequence to the performance of the disposal system.

40 CFR Part 194 specifies that mining for natural resources '*shall be assumed to occur with a one in 100 probability in each century of the regulatory time frame*' (US EPA, 1996). It also specifies in detail the assumptions and calculation process to be used for determining the likelihood and consequences of drilling events in a performance assessment:

'Inadvertent and intermittent intrusion by drilling for resources (other than those resources provided by the waste in the disposal system or engineered barriers designed to isolate such waste) is the most severe human intrusion scenario.

In performance assessments, drilling events shall be assumed to occur in the Delaware Basin at random intervals in time and space during the regulatory time frame.

The frequency of deep drilling shall be calculated in the following manner:

> *(i) Identify deep drilling that has occurred for each resource in the Delaware Basin over the past 100 years prior to the time at which a compliance application is prepared.*
> *(ii) The total rate of deep drilling shall be the sum of the rates of deep drilling for each resource.*

The frequency of shallow drilling shall be calculated in the following manner:

> *(i) Identify shallow drilling that has occurred for each resource in the Delaware Basin over the past 100 years prior to the time at which a compliance application is prepared.*
> *(ii) The total rate of shallow drilling shall be the sum of the rates of shallow drilling for each resource.*
> *(iii) In considering the historical rate of all shallow drilling, the Department may, if justified, consider only the historical rate of shallow drilling for resources of similar type and quality to those in the controlled area.*

Performance assessments shall document that in analyzing the consequences of drilling events, the Department assumed that:

(i) Future drilling practices and technology will remain consistent with practices in the Delaware Basin at the time a compliance application is prepared. Such future drilling practices shall include, but shall not be limited to: The types and amounts of drilling fluids; borehole depths, diameters, and seals; and the fraction of such boreholes that are sealed by humans; and
(ii) Natural processes will degrade or otherwise affect the capability of boreholes to transmit fluids over the regulatory time frame.

With respect to future drilling events, performance assessments need not analyze the effects of techniques used for resource recovery subsequent to the drilling of the borehole.'

4 Discussion

4.1 Regulatory Perspective

The ICRP considers it inappropriate to include the probability of future human actions in a quantitative performance assessment for comparison with dose or risk constraints. Instead, the consequences of one or more stylised scenarios should be considered to evaluate the resilience of the disposal system design to such events. While there is little scientific basis for predicting the nature or probability of future human actions, it is considered necessary to assess the consequences of such an event. Appropriate criteria for assessing the radiological impacts of human intrusion are those that would apply to the control of radiation exposures under present-day circumstances. The ICRP has recommended an annual dose of 10 mSv that may be used as a generic reference level below which intervention to control exposures is not likely to be justifiable, whilst an annual dose of around 100 mSv may be used as a generic reference level above which it should almost always be justifiable to implement measures so that exposures at this level do not occur.

Revised ICRP policy has helped to shape the UK environment agencies' approach in providing new regulatory guidance on the treatment of future human actions in the safety case. The UK regulatory guidance on the treatment of human intrusion is the most recent example of an interpretation of the ICRP recommendations, and presents a largely consistent picture with European regulatory views on this issue.

The US EPA has adopted a different approach in the regulation of the WIPP, with site-specific regulations that defined which human intrusion scenarios should be considered. The WIPP is located in a resource-rich area, so the regulations specify the procedures and assumptions that should be adopted in determining the likelihood and consequences of human intrusion.

4.2 Compliance with Regulations

A systematic approach to scenario development has been used in most performance assessments, defining an undisturbed performance scenario and alternative scenarios that include lower-probability events such as human intrusion. Future human action scenarios can be defined either by screening a comprehensive list of features, events and processes (FEPs), or through an *a priori* decision concerning potential future activities. Without a clear and traceable approach, an *a priori* decision may appear to be arbitrary. Future human actions scenarios are particularly difficult to handle in the assessment of a safety case because the range of possible future human actions is large and indeterminate, and it is not possible to determine the probability of their occurrence without recourse to largely conjectural assumptions.

Assumptions about four issues can be made to restrict the scope of an assessment without reducing the credibility of the corresponding safety case (Galson *et al*., 1999):

1. The distinction between 'recent and ongoing' activities and 'future' activities, and between 'global' and 'local' human activities.
2. Patterns of future human behaviour and potentially exposed/critical groups.
3. The timing of earliest potential intrusion.

4. The treatment of intentional intrusion.

Treatment of Recent, Ongoing, Global and Local Human Activities

Recent and ongoing human activities are those that affect an area beyond the immediate vicinity of the disposal facility and which neither the developer/operator nor the regulator can easily influence. These include global human activities such as anthropogenic climate change and local human activities that have recently taken place in the vicinity of the disposal site, such as groundwater abstraction, together with any local human activities that are certain to continue.

In contrast, future human actions are unknown activities that may take place in the vicinity of the disposal site at some time in the future and which may affect the performance of the disposal system by bypassing or affecting the characteristics of the engineered and natural barriers. Unknown future global human activities, such as nuclear war, are not considered in decisions on disposal system safety.

It is appropriate to treat recent and ongoing human activities differently to future human actions. Scenarios that include the occurrence of future human actions can only illustrate the potential behaviour of the system, whereas scenarios including recent and ongoing human activities may provide a better estimate of the undisturbed system performance than those that exclude such activities.

Future Human Behaviour and Potentially Exposed/Critical Groups

All assessments of future human actions should use present-day social structures and technological capabilities as a basis for developing scenarios to prevent undue conjecture. Assessments should be based on present-day conditions in the region of the disposal site and at similar sites. The development of similar human intrusion scenarios in several European countries and the US by either the regulators or the developers/operators provides confidence that the human intrusion scenarios listed below are appropriate for many sites:

- Exploratory drilling directly into the disposal facility, with the extraction of core samples for further analysis.

- Mining (except for sites in granite or clay as it is assumed that there is no incentive for mining at such sites at the proposed repository depths).

- Drilling of wells into the host rock for water abstraction.

- Construction of a house on excavated waste from a geological disposal facility, or disturbance of a near-surface disposal facility during subsequent construction at the site.

Some of the national differences in approaches to future human action scenarios may be explained by differences in the geographical and geological settings of the planned disposal facilities in each country, where this has been decided. Scenarios need to be developed on a site-specific basis; aspects which need to be taken into consideration when developing human intrusion scenarios include the economic potential of the host rock and the likely future population of the area. These factors will affect the likely probability of drilling, mining, water abstraction and construction activities. As the

WIPP is located in a resource-rich area, the regulators specified that mining and drilling scenarios should be considered in detail in the safety case.

The focus of assessments of future human actions should be on longer-term doses received by potentially exposed groups who might anyway be considered by undisturbed performance scenarios. In particular, human intrusion assessments should include groups considered in assessments of groundwater releases who may receive additional doses from new pathways arising from future human actions, and groups consuming foodstuffs contaminated by radionuclides brought to the surface during or subsequent to an intrusion and dispersed into the biosphere.

The Timing of Earliest Potential Intrusion

Active institutional controls can prevent or detect local disruptive activity through on-site security and surveillance, although a geological disposal facility is required to provide safety without the need for such controls in an undisturbed scenario. Institutional controls on the site of a disposal facility cannot be maintained indefinitely; however, assumptions of the time limit for effective controls vary between countries. The length of this period is important in safety assessment because the waste is most radioactive immediately after disposal.

The Treatment of Intentional Intrusion

In a deliberate intrusion scenario, the intruder is held responsible for the consequences of their actions, and all permutations of this scenario have been excluded from regulatory requirements and performance assessments.

4.3 Summary

There are some differences in approach to the regulatory treatment of future human actions between countries but, within Europe, there is good consensus on the following points:

- Actions should be taken during siting and design of disposal facilities to reduce the probability and/or consequences of potentially disruptive future human actions.
- Assessments of future human actions should be based on present-day conditions in the region of the disposal site and at similar sites. Assessments should assume present-day social structures and technological capabilities.
- Assessments of disposal system performance without disruptive future human actions should include the effects of any recent and ongoing human activities that might affect the performance of the disposal system. The potential effects of disruptive future human actions should be assessed using separate scenarios, which may be proposed by the developers/operators or the regulator. Scenario probability should only be considered qualitatively.
- Assessments should exclude consideration of deliberate intrusion, and should consider only inadvertent intrusion scenarios.

Regulatory requirements for the WIPP differ from the regulatory consensus in Europe in that the regulator defined the procedures and assumptions necessary for the developer/operator to quantitatively determine the probability of occurrence of future human intrusion scenarios.

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Part 2.2: Development of Stylised Human Intrusion Scenarios

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1 Introduction

The option favoured worldwide as final solution for the management of high-active and long-lived radioactive waste is to isolate this waste in disposal systems located in deep geological formations. The safety of such disposal systems has to be demonstrated in the so-called Safety Case. As a matter of fact, the intrinsic hazard of radioactive waste decreases with time but some hazard remains for extensive long time periods. Therefore, Safety Cases for geological disposal typically take into account demonstration periods of thousands to millions of years. Over such periods, a broad range of events and processes may have some impact on the disposal system and its environment.

For this reason, different potential evolutions of the disposal system are studied and their effects on safety are analysed. Potential evolutions, which are usually referred to as scenarios, can in principle be attributed to natural phenomena, human phenomena, as well as phenomena induced by the disposal concept and the disposed radioactive waste . It turned out that the prediction of certain events and the evolution of the society and human behaviour as well as the biosphere is rather difficult or even impossible. Especially the category of human induced phenomena including human actions at the disposal site belongs to this kind of events. However, such human actions have to be taken into account when assessing safety of the disposal system.

Future human actions involved in safety considerations in the different national approaches differ to some extent considerably from each other. The differences can be predominantly attributed to different disposal concepts, site situations, and regulations as well as country-specific perceptions. In recent years, a change of dealing with human actions from devastating human intrusion scenarios to more reasonable scenarios can be observed.

For most disposal programs the development of a common accepted technical basis concerning the treatment of future human actions is desirable. In this context, potential impairments of the isolation capacity of the disposal system through human actions, such as human intrusion into the repository, and the resulting consequences have to be discussed.

Any prediction of human intrusion over an extensive demonstration period will be speculative. It is therefore not possible to derive human intrusion scenarios systematically. As a consequence, regulatory procedures and guidance are necessary for placing and dealing with the problem. In most of the existing regulations stylisation is seen as an appropriate way for their treatment.

This report deals exclusively with the treatment of human intrusion and especially with stylised human intrusion scenarios. The report includes several chapters with the following content:

Some general information concerning this document and an introduction to the subject human intrusion are given in chapter 1. Chapter 2 addresses existing regulations, guidelines and recommendations on the treatment of human intrusion in safety evaluations. Examples of the treatment of human intrusion scenarios in safety cases are given in chapter 3. Possible approaches to the development of stylised human intrusion scenarios for a repository in a plastic clay formation and in a salt formation are presented in chapter 4. Chapter 5 summarises the essential aspects of the previous sections and chapter 6 lists the references cited in the report.

It should be mentioned here, that in the framework of the integrated project PAMINA the subject "Human Intrusion" has also been addressed from different perspectives in other reports. These are the task report “Human Intrusion” from WP1.1 in RTDC-1 and the milestone report M3.1.12 “Stylised Human Intrusion Scenarios: A Regulatory Perspective” from WP3.1 in RTDC-3. The latter is included in this report under the part 2.1.

2 Overview of recommendations and regulations on the treatment of human intrusion

Human intrusion has always been a difficult issue in performance and safety evaluations of radioactive waste repositories. It is strongly related to the two basic options that can be considered for waste management:

- concentrate the waste and isolate it from the environment;

- dilute the waste and disperse it over large volumes of the geosphere.

In the first option the radionuclides present in the waste are isolated from the human biosphere and only a small number of intruders might get in contact with the waste and incur considerable doses. In the second option the radiological consequences for intruders will be small, but large populations will come in contact with the radionuclides and incur very small additional doses. For the management of most radioactive waste types the first option is applied.

Until a few years ago human intrusion scenarios were treated in safety assessments in a similar way as scenarios induced by natural phenomena. Because the probability of occurrence of a human intrusion into a geological repository is significantly smaller than one, the risk associated with those scenarios was assessed by combining the estimated radiological consequence (dose) with the occurrence probability of the scenario. Overviews of such analyses of human intrusion scenarios can be found in NEA (1989) and Nancarrow et al. (1990).

In recent years performance assessors as well as radiological protection authorities are becoming aware that:

- human intrusion is inherent to the concentrate and isolate option;

- it is impossible to estimate probabilities of human intrusion scenarios, because of the unpredictability of the evolution of human habits and knowledge;

- protection from exposures associated with human intrusion is accomplished by reducing the possibility and likelihood of such events during site selection and repository design;

- direct exposures due to drastic intrusion scenarios are not relevant for a given site and repository concept, because they lead to the same consequences independently of the site or disposal concept.

In 1995 the National Academy of Sciences of the US formulated a number of recommendations on the treatment of human intrusion in safety assessments in a report on radiological protection standards for the proposed repository at Yucca Mountain (NAS, 1995). In the meantime most of these recommendations have been taken over in the US regulations by the Environmental Protection Agency (EPA, 2001) and the Nuclear Regulation Commission (NRC, 2001).

In 1999 the Swedish Nuclear Power Inspectorate (SKI) published a report on the consideration of future human actions in safety assessments (Wilmot et al., 1999). The International Commission on

Radiological Protection formulated a number of recommendations on the treatment of human intrusion in its Publication 81 (ICRP, 2000).

Recently, several countries, e.g. France (ASN, 2008), Germany (BMU, 2009) and UK (GRA, 2009) issued new regulations for geological disposal systems of high active waste. These regulations include also requirements for the treatment of human intrusion. In addition to that, international and national working groups, like the Franco-Belgian Working Group and the German Working Group for "Scenario Development" , published recommendations to the treatment of human intrusion (FBWG, 2004), (GWG, 2008).

2.1 Recommendations of the National Academy of Sciences of the US

In 1992 the Congress of the US directed the Environmental Protection Agency to promulgate protection standards for the proposed repository at Yucca Mountain and asked the National Academy of Sciences to advise the agency on the technical basis for such standards. The report of the National Academy of Sciences (NAS, 1995) became available in 1995. A chapter of this report deals with human intrusion and institutional controls.

The first aspect treated is the possibility to prevent human intrusion during a certain period. For an initial period institutional control can prevent human intrusions. However "there is no basis in experience for such an assumption beyond a time scale of centuries". Also on passive control such as the use of markers and archival of records the NAS report concludes: "there is no technical basis for making forecasts about the reliability of such passive institutional controls". For the estimation of probabilities of intrusion the NAS report states: "it is not technically feasible to assess the probability of human intrusion into a repository over the long term" and "it is not scientifically justified to incorporate alternative scenarios of human intrusion into a risk-based assessment". On the other hand the NAS report is much more confident in consequence analyses: "we do however conclude that it is possible to carry out calculations of the consequences for particular types of intrusion events, for example drilling one or more boreholes into and through the repository".

The NAS report identifies three categories of hazards resulting from intrusion:

- hazards to the intruders (drillers, handlers of material taken from the repository);
- hazards to the public from material brought to the surface by the intrusive activity;
- hazards to the public that arise because the integrity of the repository's engineered or geological barriers has been compromised by the intrusion.

For the first two categories of hazards the NAS report concludes "that analyzing the risks to the intrusion crew and the risks from any material brought directly to the surface as a consequence of intrusion is unlikely to provide useful information about a specific repository site or design and therefore should not provide a basis for judging the resilience of the proposed repository to intrusion. Whenever highly dangerous materials are gathered into one location and an intruder inadvertently breaks in, that intruder runs an inevitable risk". On the other hand the NAS report recommends "that the compliance analysis should concentrate on the third category of hazard posed by human intrusion, the one resulting from modification of the repository's barriers". The NAS report also describes how this last category of scenarios can be analysed: "we consider a stylized intrusion scenario consisting of one borehole of a specified diameter drilled from the surface through a canister of waste to the underlying aquifer" and "consider current drilling technology but assume sloppy practices, such as not plugging the hole carefully when abandoning it".

In the context of the considered stylised human intrusion scenario NAS holds the view that “One can always conceive of worse cases, such as multiple boreholes with each penetrating a canister, but this single-borehole scenario seems to us to hold the promise of providing considerable insight into repository performance with the minimum complication.” and further NAS recommends “Because construction of scenarios is arbitrary, we would argue for the simplest case that tests the repository.”

2.2 Rules of the Environmental Protection Agency and the Nuclear Regulation Commission in the US

In 2001 the NAS recommendations have been largely taken over by the EPA and the NRC in rules concerning the proposed repository at Yucca Mountain (EPA, 2001; NRC, 2001).

A number of important statements of the final NRC rule are given hereafter:

§ 63.321 Individual protection standard for human intrusion

DOE must determine the earliest time after disposal that the waste package would degrade sufficiently that a human intrusion could occur without recognition by the drillers. DOE must:

(a) Provide the analyses and its technical bases used to determine the time of occurrence of human intrusion(see §b 63.322) without recognition by the drillers;

(b) If complete waste package penetration is projected to occur at or before 10,000 years after disposal:

(1) Demonstrate that there is no reasonable expectation that the reasonably maximally exposed individual receives no more than an annual dose of 0.15 mSv as a result of a human intrusion, at or before 10,000 years after disposal. The analyses must include all potential environmental pathways of radionuclide transport and exposure subject to the requirements at § 63.322; and

(2) If exposures to the reasonably maximally exposed individual occur more than 10,000 years after disposal, include the results of the analysis and its bases in the environmental impact statement for Yucca Mountain as an indicator of long-term disposal system performance.

(c) Include the results of the analysis and its bases in the environmental impact statement for Yucca Mountain as an indicator of long-term disposal system performance, if the intrusion is not projected to occur before 10,000 after disposal.

§ 63.322 Human intrusion scenario

For the purpose of the analysis of human intrusion, DOE must make the following assumptions:

(a) There is a single human intrusion as a result of exploratory drilling for ground water;

(b) The intruders drill a borehole directly through a degraded waste package into the uppermost aquifer underlying the Yucca Mountain repository;

(c) The drillers use the common techniques and practices that are currently employed in exploratory drilling for groundwater in the region surrounding Yucca Mountain;

(d) Carefully sealing of the borehole does not occur, instead natural degradation processes gradually modify the borehole;

(e) No particulate waste material falls into the borehole;

(f) The exposure scenario includes only those radionuclides transported to the saturated zone by water (e.g., water enters the waste package, releases radionuclides, and transports radionuclides by way of the borehole to the saturated zone); and

(g) No releases are included which are caused by unlikely natural processes and events.

2.3 Swedish recommendations

In 1999 the Swedish Nuclear Power Inspectorate (SKI) published a report on the consideration of future human actions in safety assessments (Wilmot et al., 1999). The SKI report states, as it is the case in the US recommendations, that regulators could exclude drillers and other direct intruders from consideration in safety assessments, and that modifications of the repository system resulting from drilling should be considered. An important difference between the Swedish and US recommendations is that the SKI report recommends to consider also doses due to the dispersal of radioactive material in the biosphere as a result of drilling activities.

2.4 International Commission on Radiological Protection

In a report published in 1998, the International Commission on Radiological Protection (ICRP, 1998) agreed that the possibility of elevated exposures from human intrusion is a consequence of the decision to concentrate the waste in a repository. The Commission formulated the following recommendations:

- protection from exposures associated with human intrusion is best accomplished by efforts to reduce the possibility of such events, which may include siting a disposal facility at depth or incorporating robust design features making intrusion more difficult or employing active and passive institutional control;

- since no scientific basis exists to predict the nature and probability of future human actions, it is not appropriate to include the probabilities of such events in a quantitative safety assessment;

- because the occurrence of human intrusion cannot be ruled out, the consequences of one or more stylised intrusion scenarios should be considered by the decision-maker to evaluate the resilience of the repository to potential intrusion;

- in circumstances where human intrusion could lead to doses to those living around the site sufficiently high that intervention on current criteria would almost always be justified, reasonable efforts should be made to reduce the probability of intrusion or to limit its consequences; the Commission advises that an existing annual dose, this is the existing and persisting annual dose incurred by individuals in a given location, of around 100 mSv/a may be used as a reference level above which interventions should be considered as justifiable; similar considerations apply in situations where the thresholds for deterministic effects are exceeded;

- for situations where the total doses, i.e. the sum of the doses from background irradiation and from human intrusion, are less than 10 mSv/a, intervention is not likely to be justifiable.

For the application of the ICRP recommendations to a high-level waste repository (Cooper, 2002) reasonable effort should be made to reduce the probability of intrusion or to limit its consequences. Possible actions to reduce the probability or likelihood of intrusion are siting at depth and away from known natural resources, use of markers, maintenance of records or of other prolonged passive institutional controls. The reduction of a dose to an intruder involves dilution of the waste prior to disposal. Consequently, this is not a viable option. For doses resulting from indirect exposure of individuals in population groups criteria developed for intervention in prolonged exposure situations can be applied. In the case that the estimated total annual dose is below 10 mSv, no further action is needed. In case that the estimated annual dose is in the region where intervention might be necessary on the basis of current criteria, it should be evaluated what reasonable steps could be taken to reduce the consequences.

2.5 Franco-Belgian Working Group on the safety approach to disposal in deep geological formations

In 2004 a Franco-Belgian Working Group on the safety approach to disposal in deep geological formations published its first report (FBWG, 2004). The Working Group made a distinction between two types of consequences: (1) immediate consequences for intruders and (2) deferred consequences for individuals of a critical group associated mainly with the transfer by water resulting from a deterioration of the containment barriers by the intrusion.

In the first case, the doses received could be high and would be difficult to reduce through modification of the design of the repository. These high consequences are closely linked to the strategy of "concentration and containment" selected and comparison of the dose rate received by the intruder with a regulatory limit is not pertinent. The likelihood of the occurrence of such an intrusion can be reduced by selecting a site that is not rich in natural resources, by means of markers and by the depth of the repository.

In the second case, the situation is comparable to that of an altered evolution scenario leading to a limited disturbance of the repository. The release of activity should only affect a fraction of the repository and the radiological consequences are assessed in the general framework of the altered evolution scenarios. The design must be optimised as far as possible to reduce the consequences associated with an intrusion scenario.

2.6 Regulation in France

In 2008, a new guideline was issued in France on the disposal of high-level and long-lived intermediate-level radioactive waste. This guideline, which was developed by the French radiological protection authority Autorité de Sûreté Nucléaire (ASN), replaces the rule "Règle Fondamentale de Sûreté (RFS) III.2.f" (DSIN, 1991). ASN (2008) requires with respect to human intrusion or other future human activities the investigation of boreholes, mines and caverns as well as surface and near surface constructions. Fabrication and design defects, general defects and undetected disturbances have to be considered as well. Detailed data and descriptions of the situations which have to be taken into account are provided in an additional appendix, however without binding character. With regard to inadvertent human intrusion, it is assumed that the period for the conservation of knowledge about the repository will be 500 years. Human intrusions before this period do not have to be considered. Further, the technological state of the art is assumed to be that of today.

2.7 Regulation in Germany and recommendations

Regulation:

The “Federal Ministry of the Environment, Nature Conservation and Nuclear Safety” (BMU) issued in July 2009 the new safety requirements “Safety Requirements for the Disposal of Heat Generating Radioactive Waste” . This regulation replaces the safety criteria of 1983.

Regarding future human actions including “Human Intrusion” there are some requirements that directly or indirectly refer to this subject. These requirements comprise provisions for different topics like optimisation, risk limits, institutional control, preservation of information and knowledge and transfer of knowledge.

Requirements which relate to future human actions are:

Optimisation of the disposal system with respect to future human actions has to be considered. However, this optimisation has to be performed in a subordinated way to the other optimisation objectives, e.g. safety in the post closure phase. Since future human actions cannot be predicted, such actions have to be analysed on the basis of reference scenarios, also referred to as stylised scenarios. Normal present-day human actions have to be taken into account in the reference scenarios. This optimisation aims to reduce the possible occurrence of such future human actions and their potential radiological consequences.

For developments due to an inadvertent human intrusion into the isolating rock zone, no constraint for a reasonable risk will be defined.

As far as practically achievable, administrative provisions have to be arranged for the time after repository closure so that no human activities can take place that would jeopardise the isolation of the waste. These measures have to be conceived in such a way that they will remain effective as long as possible.

All documents containing data that might be relevant for the information of future generations have to be preserved after the closure of the repository. This includes especially information on which areas in the vicinity of the repository mine have to be protected against human intrusion and what human activities have to be subject to particular restrictions. Complete sets of documents have to be kept safe in at least two different appropriate locations.

Recommendations:

In addition, human intrusion and their possible implications have been discussed in a German Working Group on "Scenario Development". As a result, the Working Group prepared a joint position (GWG, 2008) that comprises the following recommendations how to deal with "Human Intrusion" in the safety case:

(1) The IAEA (International Atomic Energy Agency) and the OECD/NEA (Organisation for Economic Co-operation and Development/Nuclear Energy Agency) formulated the principle of concentration and isolation for the disposal of radioactive waste. On implementing this principle, the different nations pursue a strategy of disposal in deep geological formations, especially with respect to high-active and long-lived waste. In Germany, all kinds of radioactive waste are to be disposed of in deep geological formations.

(2) Disposal in deep geological formations ensures the best-possible protection of man and the environment from the harmful effects of the radioactive waste even for future generations. The objective of the decision in favour of this concept is to isolate the waste from the habitat of man and to impede the penetration to the waste considerably. However, human intrusion and thus associated potential radiological consequences cannot finally be excluded.

(3) Human intrusion (HI) is understood by the Working Group as any human activity after the closure of the repository mine that will immediately damage the barriers within the backfilled and sealed mine workings and the isolating rock zone.

(4) The Working Group holds the view that human intrusion has to be treated in the safety case.

(5) A comprehensive study of human intrusion on the basis of a systematic scenario development would require the prediction of human actions as well as of the state of the art in science and technology of future generations. The Working Group holds the view that such a prediction is infeasible. Therefore the issue of HI has to be treated apart from the systematic scenario development and thus has to be dealt with separately in the safety case.

(6) A distinction is made between inadvertent and intentional intrusion. Human intrusion is considered as inadvertent if the awareness of the repository and the knowledge of the hazard potential of the waste emplaced have been lost. In the case of intentional intrusion, society is still aware of the repository and its hazard potential. The Working Group holds the view that it is exclusively inadvertent intrusion that the safety case has to deal with. Intentional intrusion can only be placed in the responsibility of the respective acting society.

(7) The Working Group holds the view that the knowledge of the repository site and the hazard potential originated by the repository can be maintained over a period of several hundreds of years and be brought to the attention of those acting in case of any activities at the repository site. Based on documentation from German mining archives that are still in use and preserved to this day, a time span of 500 years can be assumed in this respect. The Working Group therefore recommends that inadvertent human intrusion should only be assumed to take place after at least 500 years.

(8) The Working Group holds the view that suitable and appropriate measures have to be taken upon the planning and construction of a licensed repository in the future that hinder or prevent inadvertent

human intrusion and/or reduce the consequences. These measures must not impair the safety of the repository. The Working Group holds the view that the most effective measures against inadvertent intrusion consist of establishing the repository in deep geological formations and providing knowledge maintenance in the long run. This limits the possibility of inadvertent human intrusion and the occurrence of the resulting consequences.

(9) With the decision for the concept of concentrating and isolating the radioactive waste in a repository, the possibility inevitably has to be accepted that radiation exposure limits may be exceeded in the event of intrusion into the repository. In addition, it is not possible to quantify appropriately the consequences associated with human intrusion due to the lack of predictability of the boundary conditions and other parameters to be assumed. Therefore the Working Group holds the view that it is not reasonable to evaluate consequences of HI by means of radiological limit values.

(10) Selected scenarios shall be used for the purpose of balancing measures aimed at reducing the consequences. These HI scenarios have to be derived based on the specific repository plans and site conditions. These HI scenarios need not be covering or conservative.

(11) The Working Group recommends that the spectrum of HI scenarios should be appropriately limited, e.g. for the host rock "salt" to exploratory drilling, the construction of a mine, and solution mining of caverns. The Working Group recommends that the boundary conditions for the derivation of such scenarios have to be determined on a regulatory basis, e.g., in a guideline.

2.8 Regulatory guidance in UK

A new guidance "Geological Disposal Facilities on Land for Solid Radioactive Wastes: Guidance on Requirements for Authorisation" (GRA, 2009) was issued by the Environment Agency and the Northern Ireland Environment Agency (NIEA) in February 2009. This guidance includes a specified requirement "Requirement R7: Human intrusion after the period of authorisation" with a number of paragraphs regarding the treatment of human intrusion. The content of the new guidance regarding human intrusion is presented in detail in part 2.1.

3 Examples of the treatment of human intrusion scenarios in recently published safety cases

3.1 Yucca Mountain (DoE, US)

The human intrusion scenario analysed in the Environmental Impact Statement (DoE, 2002) assumes:

- the intrusion would occur 30 000 years after repository closure, when there was enough degradation in waste packages so that the driller might not detect the penetration;
- the intrusion would result in a single, nearly vertical borehole that penetrated a waste package and extended down to the saturated zone;
- current practices for resource exploration would be used to establish properties;
- the borehole would not be adequately sealed and would permit infiltrating water and natural degradation processes to modify the borehole gradually;
- only releases through the borehole to the saturated zone were considered; hazards to the drillers or the public from material brought to the surface by the assumed intrusion were not included.

The doses resulting from the considered human intrusion were calculated probabilistically. The peak of the mean annual individual dose would be $2x10^{-8}$ Sv, occurring about 70 000 years after the intrusion.

In the more recent Supplemental Environmental Impact Statement (DoE, 2008) it is considered very unlikely that the driller would penetrate a waste package without realising it because of the presence of the drip shields and the containers. The earliest time at which a drilling intrusion could occur is estimated to be 200 000 years after closure of the repository on the basis of the fact that the waste package would be susceptible to drilling once the drip shield failed. The mean and median annual doses would be about 10^{-7} Sv and would occur approximately 2 000 years after the intrusion.

3.2 Opalinus Clay (NAGRA, Switzerland)

In the Opalinus Clay Safety Report (NAGRA, 2002) three scenarios related to future human actions are considered. These scenarios are exploratory activities, exploitation of a deep aquifer for extraction of drinking water and the abandonment of the repository before final backfilling and sealing. Among these scenarios only the exploratory activities have to be considered as human intrusions.

In the exploratory activities scenarios it is assumed that a borehole penetrates the repository. The borehole is assumed to be abandoned without sealing and without removing the casing. Initially the casing is regarded as being impermeable. As corrosion progresses, the casing becomes leaky and radionuclides migrate in the borehole. When the casing is completely corroded, the borehole collapses and finally self-seals due to convergence of the clay. The hydraulic conductivity of the borehole filled with sedimented debris of claystone was experimentally determined to be of the order of 10^{-7} to 10^{-8} m/s. In the calculations the hydraulic conductivity of the borehole ranges from 10^{-10} to 10^{-6} m/s. The transport of radionuclides through the borehole into the overlying aquifer is assumed to be instantaneous.

Three variants of borehole penetration are considered:

- the borehole penetrates a HLW (spent fuel or vitrified HLW) disposal gallery;
- the borehole penetrates a HLW canister;
- the borehole penetrates an ILW disposal gallery.

For the gallery penetration variant it is assumed that the intrusion occurs at 10 000 years shortly after breaching of the canister. For the canister penetration variant the intrusion is assumed to occur after 100 000 years, because it is not considered possible to drill through the canister walls without detection as long as the steel canister walls are at least partially intact. The intrusion into the ILW disposal gallery is assumed to occur 500 years after the end of the emplacement operations.

The highest dose is calculated for the variant considering a borehole penetration into a spent fuel canister; its value is $2x10^{-6}$ Sv/a. This value can be compared with the maximum dose calculated for the reference case, which is $5x10^{-8}$ Sv/a for the spent fuel repository. For the penetration into an ILW gallery the maximum dose is $2x10^{-7}$ Sv/a; the latter value can be compared with the maximum dose of $4x10^{-9}$ Sv/a calculated for the reference case of the ILW repository.

3.3 SR97 report (SKB, Sweden)

In its SR97 safety report SKB (1999) considers a drilling of deep boreholes scenario. It is assumed that the knowledge of the repository is lost after 300 years. Two exposure situations are considered:

- dose and risk to drilling workers that are exposed to radiation from radionuclides in the drill cuttings or in the retrieved core; both external exposure to irradiation and internal exposure due to the inhalation of dust are considered;
- dose and risk to members of a family living on the site; it is assumed that radionuclides are transported through the damaged canister and the abandoned open borehole to the surface, where the radionuclides are accumulated in a peat bog; the peat is used for cultivation of crops; consumption of those crops by man results in a dose.

By assuming an exposure time of 1 hour, the maximum dose to drilling workers is estimated to be about 20 mSv. The risk is calculated by combining the calculated doses with the probability of having a drilling through the canister.

3.4 SR-Can report (SKB, Sweden)

In the SR-Can safety report SKB (2006) has selected three scenarios related to future human actions on the basis of a technical analysis and an analysis of societal factors for the two candidate sites at Forsmark and Laxemar; at both sites the considered host formation is granite. The three selected scenarios are: an exploratory drilling through the repository, the construction of a rock cavity or tunnel in the vicinity of the repository and the construction of mine in the vicinity of the repository.

The main considered case is the drilling of an exploratory borehole penetrating a spent fuel canister 300 years after repository closure. It is assumed that cores taken from the borehole containing fragments of the disposed spent fuel are removed from the site, but cuttings that may contain fine fragments are assumed to be spread on the ground at the drilling site. In contrary to the SR97, doses to drilling workers are not calculated in the SR-Can, but only doses to a hypothetical family that is assumed to settle on the drilling site shortly after the drilling activities. The members of that family are assumed to receive doses from agricultural products from the area, from using the borehole as a water well and from direct exposure and inhalation of dust from the soil. The calculated annual effective dose from the cuttings left on the ground is considerable. The first 400 years after closure of the repository it is higher than 1 Sv. After about 15 000 years it is still 0.3 Sv/a. The doses arising from the use of the borehole as a well for drinking water and irrigation are less problematic. It is about 2×10^{-5} Sv/a 300 years after closure of the repository and it further decreases with time.

For the rock cavity or tunnel case it is assumed that an underground facility with a cross section of 100 m^2 will be constructed at a depth of 50 m above the repository. The excavated cavity or tunnel can change the hydraulic gradients in the host rock. For the Forsmark site the upper 200 m of the bedrock are more conductive than the lower part at 400 m depth. Consequently, there is no reason to expect any influence on the repository.

At the Forsmark site a rock having a potential for iron oxide mineralisation has been identified. At present, the mineral deposits are assessed to be of no economic value. Nevertheless, as this judgement may be reconsidered in the future, the exploitation of this mineralisation is addressed. It is shown that the construction of an underground facility a few hundred metres away from the repository will have a very limited pressure disturbance around the repository, and as a consequence it would not impact the safety functions of the repository.

Review of SR-Can by SKI and SSI (since 2008 the established Swedish Radiation Safety Authority (SSM) took over the responsibility and tasks from SKI and SSI):

The SR-Can safety report has been reviewed by the Swedish radiological protection authorities SKI and SSI (2008). The Swedish authorities consider SKB's method to identify human intrusion scenarios acceptable. However, they consider that the reported calculation cases are too limited and they ask to include more analyses for drilling of different repository components (canister, buffer, backfill). The consequences of an abandoned unsealed repository should also be evaluated.

3.5 NIREX studies (UK)

In its Generic post-closure Performance Assessment report NIREX (2003) considers a geotechnical worker scenario and a site occupier scenario. The dose to geotechnical workers examining samples of material taken a few centuries after waste emplacement from a ILW repository is of the order of 20 mSv/a. The dose is combined with a probability of occurrence, estimated from drilling frequencies, to calculate the risk.

NIREX published an overview report on the Treatment of Human Intrusion in Post-closure Performance Assessment (Kelly et al., 2003). A striking difference between the conclusions of this NIREX report and the recommendations and guidelines issued in most other countries and by the ICRP is that it is anticipated that quantitative evaluations of probability will remain an important part of the evaluation of human intrusion within the context of the current guidance from the radiological protection authorities in the UK.

3.6 Dossier 2005 (Andra, France)

The French regulations on geological disposal (DSIN, 1991) of 1991, which were still applicable in 2005, require the consideration of two types of human intrusion scenarios. The first one consists of a drilling crossing the repository from which cores are taken; the handling and examination of those cores can result in external exposure that has to be evaluated. The other scenario assumes that one or more boreholes have been drilled through the repository and abandoned; the borehole(s) short circuits partially or completely the host formation.

In its safety case Dossier 2005 Andra (2005) reports a number of analyses for those 2 human intrusion scenarios.

For the core handling scenario Andra assumes that cores containing fragments of the disposed radioactive waste are taken 500 years after disposal; the dimensions of the core are 1 m length and 0.1 m diameter. The dose rate due to external exposure at a distance of 0.4 m is calculated for different waste types. It is assumed that a geological worker is exposed during 10 minutes; that is the time needed to take the core out of the sampling tool and to condition it. A geological examination of the core would last much longer, but it is expected that the geologist will soon become aware of the potential danger resulting from the presence of the exotic materials in the core. The highest calculated dose is 7 mSv and is due to a core containing parts of a Ag-In-Cd control rod; these rods contain considerable amounts of 108Ag (half-life 418 years). The dose due to a core containing fragments of spent fuel is 1.3 mSv.

A comprehensive analysis of the abandoned borehole scenario has been carried out. It assumes the drilling of an exploration borehole 500 years after closure of the repository, that the borehole is abandoned and poorly sealed, that the borehole remains permanently open and that it will reach the Dogger formation, which underlies the Callovo-Oxford formation. Several variants of the scenario were analysed. For each zone of the repository (vitrified high-level waste, spent fuel, long-lived intermediate-level waste) a reference location close to a waste package was determined and variant locations were considered as well in sensitivity studies. Two variant cases were considered. In the first case it is assumed that a water well is exploited in the aquifer overlying the host clay formation at short distance from the abandoned borehole. The resulting lowering of the hydraulic head will induce a water flow from the deeper formation through the borehole that can convey radionuclides from the disposed waste to the water well. In the second case it is assumed that the borehole is drilled in an area where the hydraulic gradient over the host clay formation is upward and that the resulting water flow transfers radionuclides into the overlying aquifer. The resulting doses are then calculated as in the case of the normal evolution scenario.

4 Stylisation of human intrusion

4.1 Development of stylized human intrusion scenarios for a repository excavated in a plastic clay formation

4.1.1 Proposed methodology

As the treatment of human intrusion scenarios is part of a safety case, it can be assumed that the assessment basis has been developed and that the main input data needed for the development of human intrusion scenarios are available. These are data on the site and its geology, the waste and its physical, chemical and radiological characteristics, and the repository design.

a) Identification of relevant future human actions

Long lists of future human actions that might be considered for deriving human intrusion scenarios have been compiled (e.g. Appendix B of NEA (1995) report). Although many of these possible future human actions can drastically affect the barriers of a near surface disposal facility, they will only affect the aquifer system and or the biosphere of a geological repository. Only a limited number of human actions can affect the barriers or safety functions of a geological repository.

Future human actions that can affect the safety functions of a geological repository can be divided into two main groups of activities:

- drilling activities:
  - drilling of an exploratory borehole;
  - drilling of an exploitation borehole; examples of natural resources that might be considered are:
  - deep groundwater;
  - salt;
  - geothermal energy;
  - heat-cold storage;
  - storage of natural gas;
  - storage of CO_2;
- mining and excavation activities:
  - construction of a mine;
  - construction of tunnels:
  - waste disposal.

The first step of the proposed methodology consists in evaluating if the possible future human actions mentioned in the above list can be considered as being realistic for the considered site and its surrounding geology.

In principle, sites for which it is known that the underground contains exploitable resources will not have been retained during the site selection process. However, it cannot be ruled out that a geological layer that is not considered to be exploitable today might be considered as exploitable in the future for different economical or technological conditions, and that geological layers under the repository contain exploitable resources that have not yet been discovered today.

b) Considered intrusion time

For most human intrusion scenarios it can be expected that the radiological consequences will decrease with time. Therefore, it is important to estimate the earliest time at which intrusion has to be considered. Two aspects can be taken into consideration:

- when can inadvertent intrusion take place;
- will some of the engineered barriers hinder or prevent intrusion into the waste itself.

Inadvertent intrusion can occur when the existence of the repository has been forgotten. This time will depend on the duration of the period of institutional control and on the period during which man is aware of the existence of the closed repository. The duration of the period of institutional control is a topic that has to be discussed between the implementer and the radiological protection authorities. This period can be considered to range from one to a few centuries. After the period of institutional control, the period during which man is aware of the existence of the repository still can exclude inadvertent intrusion. This period is, e.g., determined by the availability of archives.

A number of engineered barriers can strongly hinder the drilling activities or even prevent that the drill-bit will penetrate into the disposed waste. Especially, the thick-walled metallic disposal canisters (also called overpacks or containers) that are used to pack the high-level waste can make that the drill-bit will deviate and, as a consequence, will not penetrate into the waste itself. It is also very unlikely that a thick-walled metallic canister can be perforated without being detected by the drilling team.

Barriers made of concrete can also have characteristics (for a drilling) that strongly contrast with those of the host formation. However, whether these materials will be recognised as exotic materials by the drilling team will strongly depend on the type of host formation and its characteristics.

c) Identification of exposure modes

In principle two main groups of exposure modes, resulting from a human intrusion into the repository, can be considered. In the first group it is considered that a fraction of the disposed waste is brought into the biosphere; in the second group it is assumed that the waste remains in its repository but that a number of engineered and natural barriers are short-circuited by the borehole, which can thus constitute a preferential migration pathway for the radionuclides present in the disposed waste.

In the first group of exposure modes, two types of exposure can be distinguished. The most drastic type of exposure is that man comes directly in contact with a fraction of the disposed waste. This type of exposure is characterised by intensive exposure of a small number of people during short times. An example of this type of exposure is the handling and possible the examination of a core containing fragments of the disposed waste. As well the duration of the exposure, i.e. contact time, as the nature of the examinations that are carried out will determine the radiological consequences of such drastic human intrusion scenarios. Therefore, an important element for the evaluation of the consequences of those scenarios is the time after which the presence of exotic and possibly dangerous materials in the cores will be detected by the drilling team. Another type of exposure is that small fragments of the disposed waste are present in the borehole's cuttings and that they are left on the drilling site. It can then be assumed that a few years after the drilling a small group of people settles in the drilling area and that they produce most of their food in that area and use water pumped from a shallow local aquifer as drinking water for men and cattle and as irrigation water.

For the second group of exposure modes, it is assumed that groundwater can flow through the drilled borehole (or tunnel) and come in contact with the disposed waste. The radionuclides transported by that groundwater flow can contaminate a neighbouring aquifer of which the water is drained by a local river or pumped via a water well. The amount of radionuclides that will be transported to the aquifers will depend on the dimensions and characteristics of the borehole (or tunnel) and on the components (the waste itself or a neighbouring component, e.g. the buffer) of the repository that are penetrated by the borehole.

4.1.2 Application of the proposed methodology to a repository excavated in a plastic clay formation

a) Identification of relevant future human actions

The geological formations in the area in NE Belgium that is considered for geological disposal in the Boom Clay formation do not contain known exploitable minerals. Therefore, mining activities can be considered to be unlikely. Consequently, the future human actions that have to be considered for the derivation of human intrusion scenarios are borehole drilling activities.

b) Considered intrusion time

The period of institutional control that will be applied to the Belgian geological repository has not yet been decided. It can be expected that this period will range between 100 and 300 years. The exact duration will be a topic for discussion between the implementer, i.e. the Belgian radioactive waste management agency ONDRAF/NIRAS, and the regulator, i.e. the Belgian radiological protection authority FANC/AFCN. The additional period during which the existence of the repository should be known by possible intruders can be estimated to be in the range from 200 to 500 years.

A period of 500 years (this value has also been used in French and Swiss safety cases) before which inadvertent intrusion can be considered as very unlikely seems reasonable.

The repository concepts that have been developed by ONDRAF/NIRAS are based on a number of components that are made of cementitious materials. As the Boom Clay is a plastic clay, it is necessary to use a liner to keep the galleries open during the operational period of the repository; this liner will consist of prefabricated concrete wedge blocks. The high-level waste (i.e. spent nuclear fuels and vitrified high-level waste from reprocessing) will be packed in a 3 cm thick carbon steel overpack which is surrounded by an about 60 cm thick concrete buffer; this concept is called the "supercontainer" concept. The long-lived intermediate-level and low-level waste will be repacked in 24 cm to 66 cm thick concrete containers.

The likelihood that a drilling team will recognise the presence of large amounts of concrete as exotic materials and, as a consequence, as potentially dangerous materials, has not yet been studied. In the following sections it will be conservatively assumed that the various cement based materials will not be recognised as exotic materials.

The carbon steel overpacks, which are considered to contain the high-level waste, are surrounded by the concrete buffer, which makes that the overpacks will remain during a very long period (more than 100 000 years) in alkaline conditions; these conditions result in very low anaerobic corrosion rates (of the order of 0.1 µm/a) of the carbon steel. It will thus take a few tens of thousands of years to have a significant reduction of the thickness of the overpacks. For the purpose of the present study we propose to take 50 000 years.

c) Identification of exposure modes

The types of exposure modes that have to be considered in the safety case are a topic for discussion between the implementer and the radiological protection authorities.

In Section 4.1 three types of exposure modes have been distinguished on the basis of the contact between man and the disposed waste:

- direct contact of the intruders with the waste;
- direct and indirect contact with drilling cuttings containing small fragments of disposed waste by people living on the drilling site;
- indirect contact with the waste, e.g. aquifer contamination by groundwater flow through an unsealed borehole.

1) Core handling and inspection scenarios

Although the relevance of core inspection scenarios is debatable because they are inherent to the concentrate and confine waste management strategy, the regulations in a number of countries, e.g. in France, require the analysis of this type of scenario.

Important aspects that will strongly influence the resulting dose are:

- the time of intrusion, taking account for the presence of the metallic container in the case of disposal of high-level waste; this topic has already been discussed in section b) above;
- when the core has been brought to the surface, the time after which the presence of radioactive materials in the extracted cores will be recognised; thereafter, it can be assumed that the drilling team will take precautionary measures to limit the exposure;

- volume of waste present in the core: the maximum amount of waste that can be estimated on the basis of the geometry of the disposal configuration of the repository concepts for the various waste types should be considered in the consequence analyses.

Two phases can be distinguished for the core scenario:

- core handling and transport;
- core inspection.

Each core that has been drilled will be taken out of the core sampling equipment and will be transported to the core storage. The time during which workers of the drilling team will hold the core in there hands is rather limited. We assume that it will take about 10 minutes to take the core out of the core sampler and about 5 minutes at the core storage.

Further handling of the core will normally depend upon the results of the preliminary geological examinations. At present, directly after the drilling of a geological exploration borehole, geophysical wireline measurements are carried out. The results of the geophysical wireline measurements are used to select the cores on which further examinations will be carried out. One of the standard measurements is the measurement of the natural γ-radiation. It can thus be expected that the presence of radioactive materials will be detected shortly after the completion of the borehole drilling. Consequently, inadvertent examination of the cores can be considered as unlikely.

2) Dispersal of radioactive materials scenarios

If the purpose of the geological exploration borehole is to explore geological formations that are located much deeper then the host clay formation, it is normal practice to drill the first hundreds of metres with a destructive drilling technique, i.e. without taking cores. However, it remains debatable whether the geophysical borehole logging measurements will be carried out or not.

We thus assume that the first hundreds of metres of the borehole are drilled with a destructive technique and that no geophysical borehole measurements are done at those depths. We also assume that the drilling cuttings, which might contain small fragments of disposed waste, are left on the drilling site. Some time after the drilling a small group of people settles in the area; they pump drinking water for man and cattle as well as irrigation water from a shallow aquifer and they produce most of their food on the former drilling site.

The volume of waste that will be brought to the surface by the borehole drilling activities is determined by the dimensions of the borehole and the geometry of the disposal configuration of the repository concepts for the various waste types. A typical diameter of a geological exploration borehole is 12 cm.

The repository concepts developed by ONDRAF/NIRAS are based on in-gallery disposal and they consider only one level of disposal galleries. This makes that the volume of waste that will be brought to the surface is rather limited.

For the calculation of the resulting doses a "residence post-drilling" scenario as developed in the framework of the near surface disposal programme can be used (Mallants et al., 2008).

3) Unsealed borehole scenarios

The Boom Clay formation is a plastic clay. As a consequence, if a borehole is not lined shortly after its drilling, the convergence of the clay will make that the borehole will be closed after a few weeks.

We propose to consider two variants:

- an open borehole that is closed after 2 weeks;
- a borehole that is filled with a coarse material, e.g. gravel, which hinders the convergence of the clay but that allows water to flow through the borehole.

For the first variant it is assumed that the borehole is not sealed after the geological exploration and abandoned for some reason. After about 2 weeks the geomechanical convergence of the Boom Clay will seal the borehole. However, during those 2 weeks groundwater will flow through the unsealed borehole and can come in contact with the disposed waste. The amount of radionuclides that will be transported through this preferential pathway will depend on the leaching rate and on the possible presence of an instantaneously releasable fraction. For this variant the resulting maximum concentration in aquifer can be calculated and compared with the one in the case of the reference scenario.

For the second variant it is assumed that the borehole is accidently filled with a coarse material, e.g. gravel, which hinders the sealing of the borehole by convergence of the clay. Although the hydraulic resistance of the gravel will limit the water flow, the advective velocity in the borehole will be much higher than the one in the clay formation. The amount of radionuclides that will be transported through this preferential pathway will depend on the leaching rate and on the possible presence of an instantaneously releasable fraction. For this variant the resulting maximum flux into the aquifer can be calculated and compared with the one in the case of the reference scenario.

4.2 Development of stylised human intrusion scenarios for a repository in a salt formation

4.2.1 Basis and requirements for the stylised scenarios to be defined

With the promulgation of the Safety Requirements by the BMU, there now exists in Germany the regulatory basis i. a. for the treatment of human intrusion into a repository for heat-generating radioactive waste. The Safety Requirements demand the analysis of so-called reference scenarios - which are also referred to as stylised scenarios - of an inadvertent human intrusion (HI) into the repository, on the basis of currently used techniques and common human activities.

It can be assumed that guidelines will be prepared on the basis of the Safety Requirements which will provide the applicant with the framework for the examination and assessment of reference scenarios on HI in the Safety Case. At present, no such guidelines are available yet.

Recommendations on how to treat HI in the Safety Case have already been prepared e.g. by the Working Group on "Scenario Development".

The BMU Safety Requirements and the recommendations of the Working Group on "Scenario Development" form the basis for the development of the reference scenarios presented in the following. The relevant contents of some of these framework requirements and recommendations, which have already been presented in detail in Chapter 2.7, can be briefly outlined as follows:

- The knowledge of the repository and its institutional control can be maintained for a period of 500 years.
- Human actions have to be considered in the safety analysis only after this period has elapsed.
- Only those scenarios are to be considered in which human intrusion into the repository occurs inadvertently. Deliberate intrusion is placed within the responsibility of the future acting society.
- The way in which society and human actions will develop cannot be predicted.
- Occurrence probabilities of scenarios that result from human actions can therefore be neither justified nor derived scientifically.
- In consequence of the above, the definition of reference scenarios is postulated.
- The study of human intrusion has to be done separately from the other scenarios, e.g. those induced by natural phenomena.
- The reference scenarios are to be based on today's state of the art in science and technology and on the conditions of our present-day society.
- Any consequences determined from the reference scenarios are not to be seen as limit or reference values. They are to be considered as indicators of how robust the repository system will react to human intrusion scenarios.
- In the planning and construction of a repository, suitable and adequate measures have to be taken to hamper or prevent inadvertent human intrusion and/or limit the consequences.

4.2.2 Identification of initial situations for reference scenarios

As mentioned above, the requirement for reference scenarios to be postulated derives from the notion that the development of society and human actions is unpredictable. Thus, there is no basis for deriving scientifically-founded scenarios of human action.

For the specification of reference scenarios regarding human intrusion into a repository for radioactive waste, present-day prerequisites, habits, behavioural patterns and conditions need to be considered. For example, the site-specific conditions regarding resources, the possible motives for an intrusion into respective depths, the methods and techniques developed, practical aspects, considerations regarding the efforts required and economical issues have to be taken into account or investigated. The reference scenarios thus represent possible interventions influencing the integrity of the repository under comprehensible, plausible and realistic assumptions from today's point of view; they consciously do not claim to cover all conceivable possibilities up into the area of mere speculation.

From the range of human intrusion scenarios that are discussed internationally, only those are to be considered more closely which would allow intrusion into the repository owing to the depth of the emplacement horizon (according to current conceptual deliberations approx. 500 m – 1,000 m). Other scenarios, such as

- settlement and road building,
- abstraction of drinking water or industrial water,
- direct agricultural use or
- tunnelling above a repository

are assigned to the scenario group of potential evolutions of the repository system and are to be dealt with as part of a systematic development of scenarios.

As for repositories in rock salt at corresponding depths, the following initial situations for human intrusion reference scenarios have to be discussed from the point of view of GRS, depending on the respective site conditions:

a) exploratory drilling,

b) construction of a mine,

c) extraction of geothermal energy and

d) use as storage rock (pore-space store, cavern).

The above-mentioned reference scenarios are the result of considerations related to current practices and techniques, mirrored on the properties of the host rock salt, national mineral deposits and conceptual requirements such as the depth of the repository.

Apart from the process that leads to an intrusion into the repository, essential boundary conditions for the description and analysis of the scenario in question are the area or place of an intrusion, the time of intrusion, the waste type, the kind of exposure, the exposure pathway, the group of individuals exposed, and the extent of the area affected by a possible contamination.

It has to be pointed out that the process in connection with reference scenario a) is directed at the exploration of the subsoil e.g. prospecting for petroleum, natural gas, ores, etc. The other reference scenarios (b – d), which serves for the extraction of a natural resource or of energy or for its storage, are in principle also initiated by an exploratory drilling. This means that it can be assumed that before a decision is taken especially to construct a mine and excavate a cavern for use as storage, there will be comprehensive examinations carried out in advance, i.a. in the form of exploratory drillings. The discussion or consideration of the reference scenario a) "exploratory drilling" thus gains special weight. The following remarks therefore deal - with a few exceptions - with the reference scenario a). Separate studies are required for the presentation and treatment of the other reference scenarios b) – d).

4.2.3 Identification of present-day techniques and processes

A fundamental requirement for the study of reference scenarios is that present-day techniques and processes are taken as a basis. The critical questioning of technologies used and of the accompanying investigations associated with the corresponding methods can yield information for the assessment of a possible detection of anomalies caused by the repository and the radioactive waste.

The above-mentioned reference scenarios will have to be discussed in particular for a repository in rock salt. For example, exploratory drillings will mainly be seen in connection with prospecting for petroleum, natural gas, ore or salt deposits as well as with thermal energy, all of which is directly related to the reference scenarios mine construction, geothermal energy and use as storage. However, it also has to be pointed out that exploratory drillings are also performed for scientific purposes.

The search for suitable sites for the extraction of natural resources and geothermal energy as well as the use of the storage capacity of certain rock types is associated with a high financial risk and with great effort. To minimise the risk, comprehensive investigations are carried out prior to any decisions regarding exploratory drillings with the aim to delimit areas potentially suitable for exploitation or hosting a deposit. Eventually, the exploratory drilling is to provide clarity with respect to the presence of a deposit, necessary conditions or certain characteristics.

A first step to find suitable areas is the screening of geological maps and geological explorations above ground as well as the screening of earlier deep drillings. This is accompanied by the evaluation of aerial photographs and the investigation of the soil of the region thus delimited. For the identification of favourable ground structures, different geophysical measuring methods are employed prior to an exploratory drilling, e.g. geomagnetic measurements, gravity measurements and thermal measurements as well as seismic methods.

Once an area has been delimited following preliminary investigations, an exploratory drilling has to clarify the need for further activities such as the mining, extraction and storage of natural resources.

The technical measures to be taken before, during and after the exploratory drilling are determined on the part of the commissioning party by the so-called drilling operation plan and the drilling programme and also by the relevant laws, ordinances and guidelines that need to be observed.

4.2.4 Anomalies initiated by the repository and their detection

Although the construction of a repository in deep geological formations does not leave any visible structures once all surface work has been completed, there will still be some traces left for posterity that indicate certain abnormal or unexpected characteristics of the surrounding area. Such repository-related abnormalities are:

- porosity, permeability and density differences caused by mining, opening and backfilling work,
- temperature differences for the area of heat-generating waste, and
- emplacement of dissimilar material such as containers, casks, concrete, radioactive waste, etc.

In this connection, the question arises whether the anomalies may be detected by the exploratory drilling. To answer this general question it was discussed with experts from industry and the authority, which resulted in the emergence of the following routine investigations and indicators that allow an identification of abnormalities:

- preliminary investigations to delimit areas and define boring sites (e.g. 3d seismics)
- high degree of wear of the drilling equipment (boring cutter, drill bit),
- inclination of the borehole,
- mud loss (loss of drilling fluid),
- examination of the drill cuttings, drill core and the drilling mud,
- possible exposure of the personnel (e.g. drilling staff and laboratory staff),
- investigation of the borehole environment (density, porosity, temperature, natural radioactivity, etc.) and
- investigation of the borehole geometry (inside width, slope, etc.).

The method of 3d seismics frequently employed in the preliminary investigation is in principle capable of identifying repository areas that have a low density and corresponding expansion as an anomaly if the resolution of the seismics is sufficient for the relevant depth. The detection probability of this method decreases with advancing convergence processes in the repository area.

Drilling through a layer with increased porosity generally also involves high mud loss. The probability of detecting the anomaly in the case of a backfilled emplacement drift or chamber is assessed to be low owing to the mud losses. Here, the same applies as in the case of the above-mentioned 3d seismic, method, namely that the detection probability will decrease with advancing convergence processes in the repository area.

If one assumes intact containers, i.e. containers that have not yet been substantially affected by corrosion processes, the drill bit will be either damaged beyond repair or destroyed and/or strongly deflected upon impact on the container wall made of steel and having a thickness of several centimetres. Both would be noticed immediately and would give rise to corresponding analyses.

The drilling into an emplacement drift in a backfilled area, i.e. between the containers, would probably go unnoticed if the backfill material was of the same composition and similar density and porosity as its environment.

If the ceiling above the emplacement area were lined with reinforced concrete - which has already been discussed among experts with respect to an underground marking of the repository - the same conditions would be encountered as described above regarding a drill hitting on a steel container.

It is highly probably that an analysis of drill cores would reveal any unexpected dissimilar materials from a repository. The same applies to the drill cuttings brought up to the surface by the drilling mud.

The drill cuttings accumulated at the drilling site is taken to a dump. Upon arrival at the dump, samples are taken and the mineralogical composition is analysed in the laboratory. It is seen to be very likely that any contaminated material that has not been detected so far will then be identified. The same applies to the disposal of the drilling mud.

There is an extremely high probability that any abnormalities will be detected in the investigations of the borehole environment that are generally carried out after drillings, using measuring sensors. For example, any radioactive radiation would be identified by the gamma log that is part of the standard measurements. In the same way, temperatures that are higher compared with normal ambient temperatures - which would be caused by the heat-generating waste - would also be noticed. Other standard methods, such as density log, sonic log and resistance log, can contribute to the detection of anomalies or unexpected properties.

The final conclusion of the discussion was that with the large number of investigations and the many different kinds of measuring methods, it would be highly likely that any anomalies connected with the presence of a repository would be noticed within the framework of the exploratory drilling.

4.2.5 Consideration of different cases

The analysis of the reference scenario a) "exploratory drilling" has to be done with consideration of the initial basis, such as regulatory requirements and guidelines, specific site characteristics, and the underlying repository concept. For such a generic analysis, the following boundary conditions have been specified:

- The human intrusion takes place 500 years after the closure of the repository at the earliest.
- Depending on the convergence of the host rock and the kind of backfill used, the pore volumes in the backfilled chambers and drifts have not yet reduced to the intact rock pore volume.
- The corrosion of the metal containers has to be considered time-dependent.
- The temperature development due to the heat-generating waste has to be considered.

In the opinion of GRS, the following different cases are relevant for the analysis of the reference scenario a) "exploratory drilling" in connection with an assumed drift emplacement:

- Exploratory drilling through a final storage container holding spent fuel assemblies located in a backfilled emplacement drift.
- Exploratory drilling through a HLW package located in a backfilled emplacement drift.
- Exploratory drilling through a backfilled emplacement drift between spent fuel containers or HLW packages.
- Exploratory drilling through an ILW package located in a backfilled chamber.
- Exploratory drilling through a backfilled ILW chamber.
- Exploratory drilling through a backfilled drift without radioactive waste.

- Exploratory drilling through the isolating rock zone outside the mine area.

It has to be pointed out that the penetration of a drill through a waste package is only possible if the metal containers are thin-walled or have corroded to such an extent that there is no noticeable resistance to the drilling bit/boring cutter. Otherwise, it is not possible to drill through a waste container as the drill would either be deflected or show strong signs of wear, which would lead to the drilling being stopped or abandoned.

Taking into account the possible ways of detecting abnormalities presented in Chapter 4.2.4, the probability of detecting the different cases a1) to a7) is assessed as follows:

The probability of a detection of abnormalities is highest for cases a1) to a3) and lowest for cases a6) and a7).

The detection probability of abnormalities is to be rated as high for case a4); for case a5), it is to be considered as given under certain constellations (e.g. containers are damaged or destroyed and the waste is spread through the backfill material).

Furthermore, certain parameters may be applied that narrow down the periods of the detection of certain abnormalities, such as the consideration of the life time of containers and waste-related temperature developments.

If abnormalities are detected, the assumption is that the case is investigated and corresponding security measures are taken.

If no abnormalities are detected, this may in principle have two reasons:

- At the time of the exploratory drilling, there are no detectable abnormalities, taking the different cases a1) – a7) into account.
- Present abnormalities are not noticed, e.g. due to human error and/or technical failure.

Irrespective of the reasons mentioned above, there are two general cases to be considered in the further analysis:

- Investigation of the influence of further undertakings in the sense of the reference scenarios b) – d) or of production drilling (e.g. petroleum or natural gas) based on the positive result of the exploratory drilling
- Investigation of the influence of the unsealed/sealed exploratory drilling

Finally, the following need to be mentioned:

It is obvious that a close cooperation between regulators and implementers on the treatment of HI in the safety case is needed. Therefore, reference scenarios of human intrusion should be provided in agreement of both parties based on comprehensive and thoroughly discussions.

Reference scenarios should generally be subjected to a qualitative analysis regarding a possible detectability of abnormalities, applying present-day techniques and procedures.

Furthermore, a distinction has to be made between the different cases of a reference scenario, such as different courses of exploratory drillings in the repository, taking concrete repository plans and site conditions into account.

GRS recommends not to determine the exposure of intruders, laboratory and operating personnel resulting from contaminated material that is transported above ground, from the examination of this material on site and/or at the laboratory as well as from its transport to dumps as the quantification of such radiological consequences is not possible with due accuracy.

However, those cases that suggest a low detection probability do have to be analysed in particular with respect to their long-term effects regarding possible radiological consequences for the population and the environment.

Any identified consequences should not be measured on limit values but should be seen as indicators for assessing the robustness of the disposal system.

5 Summary

The safety of disposal systems for radioactive waste has to be demonstrated in the safety case. In this context, one of the essential elements is the development of scenarios illustrating how such disposal systems might evolve in the course of the envisaged demonstration period. However, the evolution of some boundary conditions of the repository system like the biosphere and the society, inclusive human actions and especially human intrusion into the disposal system, is highly uncertain. As a matter of fact, the evolution of those boundary conditions cannot be predicted with the necessary accuracy. Furthermore, they cannot be derived in a systematic way like other processes and events from natural or waste and repository induced phenomena. Nevertheless, adverse effects of such boundary conditions on the disposal system cannot be excluded and have to be considered in the safety case. For many countries it seems to be an appropriate way to provide stylised scenarios for their treatment. This report deals exclusively with stylised human intrusion scenarios developed within WP 3.1 of the integrated project PAMINA.

Due to the fact that any prediction of human intrusion over an extensive demonstration period will be speculative, it is expected that regulatory procedures and guidance will be provided for dealing with the problem.

Since 1995 national as well as international radiological protection authorities are becoming aware that human intrusion scenarios have to be treated in a different way than scenarios resulting e.g. from natural events. Safety assessments carried out before 1995 were often strongly focusing upon disruptive scenarios that can lead to considerable doses to a small number of inadvertent intruders. An example of such a scenario is the examination of a contaminated core by geological workers. However, those high doses are not relevant for a given site or repository concept.

Recently the radiological protection authorities recommends to pay more attention to scenarios in which the main barriers of the repository are disrupted by a geological drilling and in which the borehole constitutes a possible pathway for accelerated transport of radionuclides. Those analyses should focus on the ability of the affected repository system to perform its main functions.

The estimation of probabilities of human intrusions has been abandoned. However, the applicability of various actions to reduce the likelihood of intrusion should be considered. Such actions are siting at depth and away from known natural resources, maintenance of records or of other prolonged passive institutional controls.

Examples from several safety cases performed by different national organisations show that human intrusion scenarios were analysed separately from the other scenarios and occurrence probabilities of human intrusions were not estimated.

The time after which a human intrusion is no longer ruled out ranges between 300 (SKB) and 500 years (Andra). In various safety cases it is assumed that the thick metallic container will considerably hinder the drilling and that, as a consequence, perforation of HLW containers will not occur during the first 30 000 years (DoE, 2002), 100 000 years (NAGRA, 2002) and even 200 000 years (DoE, 2008).

The examples of human intrusion scenarios considered in the various safety cases show also that the applied scenarios differ to some extent widely from each other, depending on the disposal concept, national regulations and guidelines. Some safety cases analyse a dose to an intruder or geotechnical worker who comes in contact with cores containing fragments of the disposed waste. Other safety cases consider exposure of people living in the neighbourhood of the geological repository after the drilling of a borehole from which cuttings containing small fragments of the disposed waste remain on the drilling site. Indirect exposure is considered in most safety cases, i.e. groundwater flows through the borehole drilled through the repository and by-passes the natural barrier provided by the host formation. For this scenario different locations at which the repository is hit by the drilling can be considered; i.e. the borehole can penetrate into the buffer, but as an extreme case it can directly come in contact with the disposed waste.

The main focus of this report relates to the development of stylised human intrusion scenarios for a repository of high active waste in a plastic clay formation and in a salt formation. This work was done by SCK•CEN for clay and GRS for salt independently from each other. Both contributions comprise a detailed description of what has been taken into account for the development of stylised human intrusion scenarios.

A methodology has been developed for the clay formation that allows the identification of a set of stylized human intrusion scenarios. The methodology is based on a systematic analysis consisting of 3 steps: identification of the relevant human actions taking into account the considered host formation and disposal site, considerations about the intrusion time and the identification of possible exposure modes. The proposed methodology has then been applied to the case of disposal in a plastic clay formation.

In case of the salt formation the regulatory framework and some recommendations used as the basis for the development of stylised human intrusion scenarios were presented. Relevant initial situations or actions as a potential basis for stylised human intrusion scenarios were identified and discussed. These situations and actions have to be investigated under the assumption of current technology and knowledge. Therefore, common techniques and procedures e.g. in the context of an exploratory drilling were discussed. A list of indicators was identified that might serve as a basis for the evaluation of the detection probability of anomalies associated with the disposal system and the emplaced radioactive waste. Several cases of exploratory drillings penetrating different locations of the disposal system were analysed regarding their detection probabilities.

Finally, it should be noted that the approaches for the development of stylised human intrusion scenarios for both in clay and in salt are more or less in good accordance with each other. Furthermore, both studies came to the following conclusion:

It is evident that the treatment of human intrusion in a safety case is necessarily based on a large number of assumptions and, consequently, it requires a number of interactions between the implementer and the regulator. These interactions between the regulator and the implementer should serve as a basis for the development of guidelines.

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Part 2.3: Stylisation of Scenarios with Very Low Probability

(Prepared by A. Vokál, NRI, Czech Republic)

1 Introduction

This report serves for developing a methodology of derivation of stylised scenarios leading to the release of radionuclides into the environment from disposal of spent fuel assemblies in carbon steel canisters surrounded by compacted bentonite in granite host rock initiated by events with very low probability. In this work stylised scenarios are not limited only to human intrusion scenario, but are related to all the scenarios initiated by very low probability events, for which it is not possible to quantify probability.

In the Czech Republic the probability of naturally occurring disruptive events is minimised for example by the following exclusion criteria given in the decree of the Czech regulator on siting of nuclear facilities:

- Any manifestation of post-volcanic activity, such as the escape of gases, or the occurrence of thermal or mineralised waters, found on the land of the proposed site or in its vicinity.
- 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 land of the proposed site.
- The occurrence of capable and seismogenic faults with recent surface deformation and with the possibility of secondary faults, found by geological survey on the land of the proposed site.
- The probability of future human activities on the site of a DGR is also reduced by the following exclusion criteria:
- The existence of a significant underground water supply or mineral waters in the site vicinity.
- The occurrence of minable raw materials in the site vicinity.

The probability of events such as critical reactions or sudden degradation of engineered barriers (e.g. premature failure of canisters) due to some human error in development and construction facility is limited to minimum by strict QA requirements on all the development and construction activities.

The probability of all these events is therefore almost negligible, but it cannot be said that it is zero due to very long time scales required for safety assessments of repository. The stylised scenarios can help to quantify possible consequences of scenarios initiated by these low probability events without considering the probability of their occurrence. In this case, however, it is not possible to quantify the potential risk from stylised scenarios, because the probability of these events is not known. Even exceeding the effective dose for an individual from a group of population for these scenarios should not always mean that a concept or site for a repository is not suitable, but could remind that it could be useful to try to quantify the probability of initiating event through systematic research or through formal expert judgment methodology.

In our work the development of stylised scenarios is inherent part of scenario derivation based on analyses of repository safety functions. The development of stylised scenarios is in this work specific to Czech concept of disposal of spent fuel assemblies in carbon steel canister surrounded by compacted bentonite in granite host rock.

2 Safety function analyses of a disposal system

In the contributions given in work package 1.1 of PAMINA project, three main categories of safety functions were distinguished for underground disposal system.

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

It was shown, however, that in various EU countries, these functions can have slightly different meaning and can be allocated on different components. For example containment function (called “isolation” by SKB and POSIVA) is in Swedish, Finish or Belgian concept provided by metallic canister, but in Germany concept in salt by the host rock formation itself. These safety functions have also in different concept various importance. While in Swedish of Finish concept, the most important safety function is containment by corrosion resistant copper canisters, in Belgian concept, it is “limited and delayed releases” function provided by slow movement of radionuclides through clay rock.

Czech concept is based on disposal of spent fuel assemblies in carbon steel canister surrounded by compacted bentonite in granite host rock. It differs significantly from Swedish or Finish concept by the fact that carbon steel canister covered by thin corrosion resistant Ni alloy could provide a good containment, but certainly not so efficient as thermodynamically stable copper based canisters. The same can be said about granite, which can also provide an excellent limited and delay release of radionuclides to the environment, but certainly not so good as almost impermeable clay. In the case of Czech concept therefore both safety functions have the same importance.

The development of safety functions in Czech DGR programme was significantly affected by systems engineered approach, in which function analysis covers identification of scenarios leading to possible failure of a system and structuring them to identify the main functions of complex system under development. The advantage of this approach is that the proposed disposal system is analysed strictly in top down direction with clear relation to the main objectives of system.

In agreement with this approach we started our function analyses with formulation of main objectives of underground disposal according to IAEA Safety Standards (IAEA, 1989): *to isolate high level wastes from the human environment and to ensure the long-term radiological protection of humans and the environment.* It is further said in these standards 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. Gradual processes are considered to include all evolutionary processes affecting the disposal. Disruptive processes are those processes that occur as random events and may have a disruptive effect on the repository and its environment.

The whole system leading to the underground disposal of spent fuels assemblies and other high-level waste was divided into the following 3 subsystems (Fig. 1):

- Disposal system of SF/carbon steel/bentonite/granite concept (DGR)
- Environment system

- DGR development and construction activities system

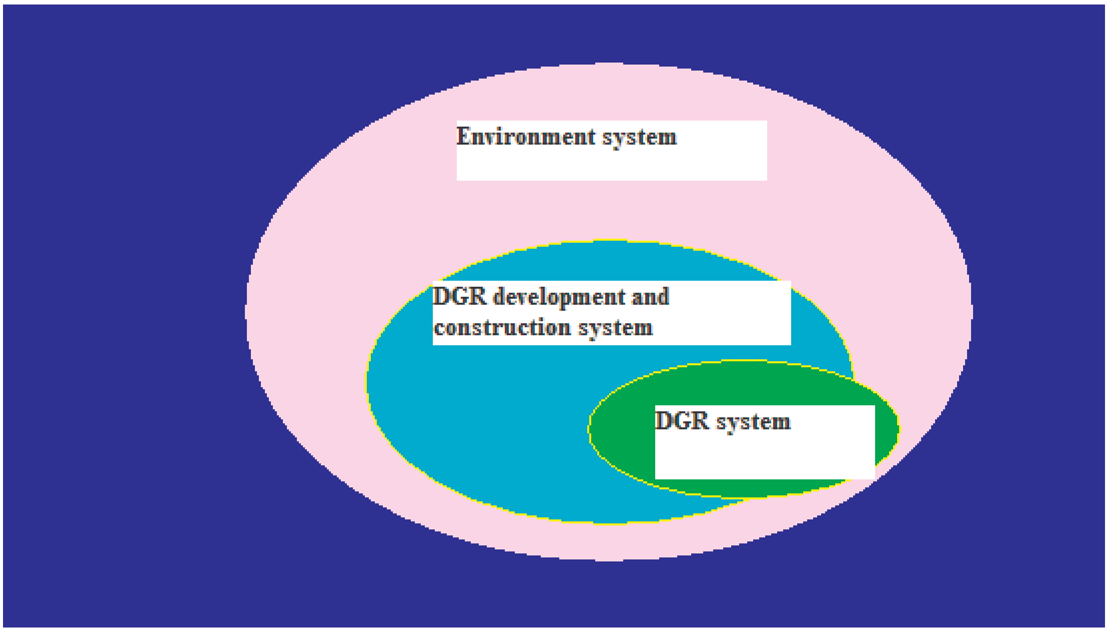

Fig. 1: DGR and surrounding systems

These three subsystems should guaranty that the main objectives of underground disposal will be fulfilled.

The main safety function of DGR in agreement with WP 1.1 conclusions can be formulated as follows:

Limit releases of radionuclides to the environment by containment and delay

This safety function cover at this level both “containment” and “limit and delay” functions identified in WP1.1.

A DGR will work only under some assumptions concerning the quality of the environment and development and construction activities. These assumptions can be transferred to safety functions allocated on environment (site) and development and construction activities, for example as follows:

Environment system (site) will provide favourable conditions for DGR

Development and construction activities will provide a robust system, which will not be sensitive to human errors and ensure minimum errors through QA system.

From comparison of above-mentioned three safety functions identified in WP1.1, it follows that these 2 functions can be identified with the stability/isolation function identified in WP 1.1. They can be further decomposed through detailed analyses of Features, Events and Processes in the agreement structuring of international NEA/OECD database of FEPs (OECD, 2000) as follows:

Environmental effects

- Geological processes and events – processes arising from the wider geological setting and long-term processes
- Climatic processes and events – processes related to global climate change and consequent regional effects;
- Future human actions – human actions and regional practises in the post-closure period that can potentially affect the performance of the engineered and/or geological barriers (not intrusive actions)

Development and construction activities effects

- Repository issues – decisions on design and waste allocation, and operations and closure and issues affecting initial state of components
- Events related to site investigation
- Operations and closure and issues affecting initial state of components

The first level safety functions of underground disposal system are summarised in the table 1.

Tab. 1: First level safety functions for SF/ carbon steel/ bentonite/ granite concept

ID	Function title	Allocated to	PAMINA/WP1 category
1	Limit releases of radionuclides to the environment by containment and delay	Disposal system domain (DGR)	Containment and Limit and delay functions
2	Provide favourable conditions for DGR	Environment	Stability/isolation
3	Provide robust system not sensitive to human errors	Development and construction activities	Stability/isolation

At this first level, a disposal system is considered as a black box characterised by some global parameters, such as inventory of radionuclides or dimensions.

3 Derivation of main categories of scenarios leading to the failure of disposal system

It is supposed that under normal evolution of repository (normal evolution scenario), a DGR will behave as designed. But it is not clear what is normal evolution scenario of DGR, because it depends on conditions given by the environment and development and construction activities and also what are “designed” properties of DGR. The following basic assumptions have been accepted for Czech disposal system covering also the site and development and construction activities.

1) The environment conditions in future years will correspond to natural changes in a site corresponding to changes in last 100 000 years (earthquakes intensity, climate change, etc) and no human intrusion will occur under normal repository evolution. It is also specified by site exclusion criteria given above.

2) No errors of disposal system will occur due to the development, design, construction or component manufacture error more than “designed” (sealing system, canister premature failures, etc) will occur under development, construction and operational phases of the disposal system. It will be ensured by careful QA system.

3) “Designed” properties of repository for normal evolution scenario will be specified in further stages of repository development.

These basic assumptions form a baseline for normal evolution scenario. The failure of the assumptions should be normally characterised by the probability of and initiating event and time of this event leading to the failure of “designed” properties of DGR. In the case, where it is not possible to quantify probability or to determine the time of the event, these scenarios can be evaluated as stylised scenarios. The basic 5 categories of scenarios derived from analyses of External FEPs from OECD/NEA database are given in table 2.

The external factors taken from the international or national databases will be then judged according to the following criteria to specify possible initiating events:

- Relevancy to
    - Repository design (geology, selected barriers)
    - Waste
    - Geographic setting
- Probability of occurrence in given time of assessment (if it is not possible to quantify the probability and it is not possible to say that it is zero, then scenarios can be evaluated as stylised scenario)
- Consequence of the external effect (e.g. negligible, low, medium, high, very high with probable time of occurrence if it is possible)

The stylised scenarios for human intrusion are dealt in other part of the project. This report is focused therefore on the development of scenarios leading to the release of radionuclides due to other causes.

Tab. 2: Basic categories of scenarios derived from analyses of external FEPs of OECD-NEA database

Scenario category number	Title	Description
1	Normal evolution scenario	Climatic changes and natural and human errors and activities has no significant effect on repository under normal evolution scenario of repository, it is not supposed any deviation from designed properties of system components
2	Geological processes and events scenarios	A geological event or process will affect unfavourably disposal system (e.g. preferential path, premature failure of engineered barrier) (OECD/NEA - FEPS code 1.2)
3	Climate change scenarios	Climatic changes will lead to significant change of conditions, leading to the conditions in a repository (OECD/NEA - FEPS code 1.3)
4	Repository issues scenarios	Human error relating to development and construction activities will lead to a significant change of conditions and premature failure of components, preferential paths, criticality etc., under this scenario can be also considered insufficient knowledge of processes occurring in a repository or site. For example badly characterised site can lead to some fast preferential paths leading from a repository to the environment. (OECD/NEA - FEPS code 1.1)
5	Future human actions scenarios	Human intrusion can lead to significant change of conditions, failure of barriers or direct contamination of people (OECD/NEA - FEPS code 1.4)

4 Scenario specification using safety function analyses

The scenarios are characterised by

a) Initiating events or processes (FEPs) (the processes like corrosion of canisters, their failure due to pressure, leaching of radionuclides from waste form or migration through engineered and natural barriers are considered as processes initiated scenario, because they are taking into account in designed properties of the repository).
b) Consequences for disposal domain subsystems and component.

Consequences of scenarios can be specified using the impact of initiating events on safety functions of disposal subsystems and components.

At the second level of DGR decomposition there are 3 subsystems:

- Repository
- Host rock
- Biosphere

There is no safety function for the biosphere due to the susceptibility of the biosphere to change on a horizon of million of years. The main safety functions of disposal system domain in relation to release of radionuclides to the environment are:

1.1 Limit releases of radionuclides to geosphere by containment and delay - repository

1.2 Limit release of radionuclides to biosphere by delay, dispersion and dilution - host rock

Repository was in previous chapters represented as a black box of certain dimensions covering both the waste and all the engineered materials introduced in the host rock system by man. Boundary between the repository and host rock are at the interface between buffer/backfill and undisturbed host rock (EDZ). At the third level of system decomposition, a repository is divided into:

- Various waste forms (spent fuel matrix, glass matrix, structure materials, instant release fraction, etc.
- Canisters
- Buffer
- Other engineered structures

In reality a repository in a granite host rock will be composed of a large number of disposal units with different properties. Contrary to other host rock like clay it is very difficult to average properties of all disposal units due to high heterogeneity of granite host rock. Each disposal unit or group of disposal units will be strongly affected by host rock conditions, primary by flux of water contacting disposal units. It is probable that some of disposal units will have to be located in more fractured areas, which will be less favourable for fulfilling safety function of repository than other. But it is difficult to differentiate all disposal units and therefore some more precise differentiation will have to be carried out only after more detailed characterization of site. Here we consider some average properties of subsystems and components without taking into account the heterogeneity of host rock.

Primary safety functions of repository (in relation to primary safety function or repository 1.1) can be formulated as follows (table 5). No primary safety function is allocated on other engineered structures.

Tab. 3: Primary safety functions of repository

ID function	Function description	Allocated to
1.1.1	Limit leaching of radionuclides from waste forms	Waste form
1.1.2	Isolate radionuclides as long as possible	Canister
1.1.3	Delay release of radionuclides to geosphere	Buffer

Granite host rock can be divided into the following main components

- Intact host rock surrounding disposal units (IHR
- Hydraulically active zone (HAZ)

The primary safety functions of host rock can be formulated as in the table 4.

Tab. 4: Primary safety functions of host rock

ID function	Function description	Allocated to
1.2.1	Limit and delay release of radionuclides to hydraulically active zone by very low advection, matrix diffusion, sorption and dispersion	IHR
1.2.2	Limit and delay release of radionuclides to the environment by low advection, matrix diffusion, sorption, dispersion and dilution	HAZ

Both host rock and repository must fulfil also secondary safety functions, which are needed for fulfilling primary safety functions of repository and host rock, which can be fulfilled only under some conditions fulfilled by repository and host rock. These secondary safety functions can be formulated as follows:

1.3 Host rock will provide favourable conditions for repository safety functions

1.4 Repository will provide favourable conditions for host rock safety functions

Identification of favourable conditions can be achieved through analyses of all the interactions between repository and host rock and biosphere using international and national databases of FEPs for disposal System Domain in NEA-OECD database (OECD, 2000), where the following items were identified:

1) Mechanical processes and conditions
2) Hydraulic/hydrogeological processes and conditions
3) Thermal processes and conditions
4) Chemical/geochemical processes and conditions
5) Biological/biochemical processes and conditions
6) Gas sources and effects
7) Radiation effects

8) Nuclear criticality

On the basis of FEPs analyses further (daughter) safety functions of host rock and repository were identified (Table 5 and 6)

Tab. 5: Secondary safety functions for host rock

ID	Function Description	Allocated to
1.3.1	Provide favourable mechanical conditions for repository	Host rock mechanical properties
1.3.2	Provide favourable chemical conditions for repository	Host rock chemistry
1.3.3	Provide favourable thermal conditions for repository	Host rock thermal properties
1.3.4	Provide favourable hydrogeological conditions for repository	Host rock hydro properties
1.3.5	Not affect unfavourably host rock by repository gases and repository itself for repository	Host rock permeability for gases
1.3.6	Provide favourable microbial activities	Host rock microbiology
1.3.7	Provide conditions protecting against possibility of critical nuclear reactions in host rock or repository itself	No clear

Tab. 6: Secondary safety functions for repository

ID function	Function description	Allocated to
1.4.1	Prevent mechanical failure of repository/canisters	Repository mechanical properties
1.4.2	Provide low flux of water contacting repository and host rock itself	Repository hydro properties
1.4.3	Provide favourable chemical conditions for low release fraction of radionuclides from a repository and host rock itself	Repository chemical properties
1.4.4	Provide favourable thermal properties for heat transfer from a repository and host rock itself	Repository thermal properties
1.4.5	Provide favourable microbiological properties for repository safety functions and host rock itself	Repository microbiology
1.4.6	Provide sufficient transfer of gases form repository and host rock itself	Repository permeability to gases
1.4.7	Provide conditions unfavourable for critical reactions and host rock itself	Repository properties relevant to criticality

The effect of an initiating event on these safety functions can specify a scenario. An example of the specification of a scenario initiated by earthquake is given in the table 7.

Tab. 7: Example of scenario component derived from analyses of interactions between repository and host rock initiated by earthquake

Scenario component ID	Allocated to	Description	Screening
2_1.1.1	Waste form	After canister failure waste form could be damaged by earthquake	Yes
2_1.1.2	Canister	Sudden rupture canisters due to earthquake can lead to immediate failure of higher fraction of canisters then expected in normal evolution scenario	Yes
2_1.1.3	Buffer	No change, It is supposed that any possible fracture will be quickly healed by bentonite swelling. This further identifies, however, new safety functions for bentonite	No
2_1.2.1	Intact host rock	Increase of hydraulic conductivity due to the formation of fractures can lead to faster transport of radionuclides	Yes
2_1.2.2	Hydraulically active zone	Increase of hydraulic conductivity due to the formation of fractures	Yes
2_1.3.1	Host rock mechanical conditions for repository	Possibly destruction of some part of rock supporting mechanically repository	Not clear
2_1.3.2	Host rock chemical conditions	Earthquake can lead to the change of chemical conditions outside range of normal evolution conditions	Yes
2_1.3.3	Host rock thermal conditions	There is no reason for thermal changes due to earthquake	No
2_1.3.4	Host rock hydraulic conditions	Higher flux of water through repository will cause greater degradation of engineered barriers	Yes
2_1.3.5	Host rock permeability for gases	There is no reason for unfavourable impact of gases	No
2_1.3.6	Host rock microbiology	There is no reason for unfavourable impact of microbes	No
2_1.3.7	Host rock properties	Earthquake could possibly cause sudden	No clear

	relevant to critical reaction	change of conditions leading possibly to criticality reaction	
2_1.4.1	Repository mechanical properties	Earthquake can cause possibly some changes in mechanical properties of repository	Not clear
2_1.4.2	Repository hydro properties	Earthquake can cause possibly some changes hydrogeological properties	Not clear
2_1.4.3	Repository chemical properties	No change	No
2_1.4.4	Repository thermal properties	No change	No
2_1.4.5	Repository microbiology	No change	
2_1.4.6	Repository permability to gases	Can possibly to cause immediate transfer of gases to the environment	Yes
2_1.4.6	Repository properties relevant to criticality	Earthquake can possibly to change immediately the conditions relevant to criticality	No clear

In the same way the effect of other categories of scenarios will be judged. Other examples will be given in further phases of the project. Scenarios will be specified by allocating values for parameters changed by an initiating event or process contrary to normal evolution scenario. For example in normal evaluation scenario, it is considered that no more that 2 canisters can fail in repository per year due to corrosion process. In the case of earthquake, much higher fraction of failed canisters will fail in one year will be considered. If the probability of earthquake and their frequency can be estimated, it can be calculated as altered scenario, but if the probability is very low and cannot be quantified then it is stylised scenario.

This is very similar to what-if scenarios with the one exception that what-if scenarios are often very unrealistic, but stylised scenarios are realistic, but with very low probability.

5 Conclusion

This milestone report summarised the main principles of the identification and structuring of FEPs and scenarios and components of DGR in evaluation of safety of deep geological repository using systems engineering approach for spent fuel/carbon steel/ bentonite/ granite concept. The stylised scenarios are here understood as scenarios initiated by a very low probability event for which it is not possible to quantify probability of occurrence.

The approach is still under work and reflects a great difficulty in achieving the simple, logic approach for derivation of scenarios initiated by events with very low probability of occurrence.

6 Literature

International Atomic Energy Agency, „ Safety principles and Technical Criteria for the Underground Disposal of High Level Radioactive Wastes," *IAEA Safety Standards Safety Series No. 99*, 1989

NEA/OECD, Features, Events and Processes (FEPs) for Geological Disposal of Radioactive Waste, OECD 2000