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H12: Examination of safety assessment aims, procedures and results from a wider perspective

April, 2004

National Cooperative for the Disposal of Radioactive Waste

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H12: Examination of safety assessment aims, procedures and results from a wider perspective

Fiona. B. Neall* and Paul.A. Smith** (Editors)

Abstract

Safety assessments (SA) are a familiar tool for the evaluation of disposal concepts for radioactive waste. There is, however, often confusion in the wider community about the aims, methods and results used in SA.

This report aims to present the H12 SA in a way that makes the assessment process clearer and the implications of the results more meaningful both to workers within the SA field and to a wider technical audience. The reasonableness of the assessment results, the quality of the models and databases and redundancy within the natural and engineered barrier system have been considered.

A number of recent and somewhat older SAs that address a range of different waste types, host rocks and disposal concepts have been considered, and comparisons made to H12.

A further aim is to put both doses and timescales in a more meaningful context. It has been necessary to:

- consider ways of demonstrating the meaningfulness of calculations that give results for many thousands of years in the future;
- provide a framework timescale as a context for SA results over long times;
- demonstrate the smallness of the risk associated with the doses by comparison with other radiological and non-radiological risks.

The perception of risk, which is a critical issue for public acceptance of radioactive waste disposal and must be considered when seeking to present safety assessment results 'in perspective' to a wider audience, is also discussed.

It is concluded that H12 is comparable in many ways to assessments carried out internationally. Some assumptions are somewhat arbitrary reflecting the generic stage of the Japanese programme, and are likely to become better founded in future exercises. Nevertheless, H12 provides a clear and well-founded message that it is feasible to site and construct a safe repository from HLW in Japan

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第２次取りまとめの安全評価の目的、手法、結果に対する理解を深めるための考察

F. B. Neall* P.A. Smith**

要　旨

放射性廃棄物の処分概念の評価手法として、安全評価は広く用いられている。しかしながら安全評価の目的、適用される方法、さらには結果については、必ずしも他の科学技術領域を含む幅広い分野の専門家間で共有されているものではない。

本報告書の目的は、安全評価に関わる研究者と幅広い技術分野の方々の双方に対し、評価の過程をより明確にし、種々の結果の関連についての理解を深められるよう、第２次取りまとめの安全評価を示すところにある。ここで、評価結果の妥当性、モデル及びデータベースの品質、並びに人工バリアと天然バリアの組み合わせによる機能の重複性を検討対象とした。

また、高レベル廃棄物以外の廃棄体やわが国とは異なる母岩、処分概念を対象としている他国の安全評価の事例について、第２次取りまとめとの比較を行った。

さらに、安全評価における線量結果及び時間スケールへの対応について、より理解を促進しうる文脈にて表すことに主眼を置いている。このため、下記の事項が求められた。

・将来何千年もの期間にわたる計算結果について、その理解を深めるように示す方法を検討すること
・長期にわたる安全評価の結果の説明のための枠組みとなる時間スケールを与えること
・他の放射線リスクや非放射線リスクとの比較により、線量に関連するリスクの小ささを明示すること

リスクの認知についても検討した。これは、放射性廃棄物処分の公衆の受容に対する重要課題であり、安全評価の結果を正しく、より広い聴衆に対して説明する際に考慮しなければならないものである。

結論として、第２次取りまとめは、他の各国にて実施された安全評価に対し、多くの点で比肩しうるものであると言うことができる。なお、日本ではサイト・ジェネリックな段階であることから、第２次取りまとめで用いられている仮定のいくつかはある程度任意性があることを否定できないが、将来の活動に伴い、より根拠の確かなものになることが考えられる。

　いずれにせよ第２次取りまとめは、日本における高レベル放射性廃棄物の安全な処分に向けてサイト選定及び処分場建設が実現可能であるという、明確かつ根拠のあるメッセージを提示している。

本報告書は、スイス放射性廃棄物管理協同組合と核燃料サイクル開発機構との放射性廃棄物管理分野における取り決めに基づく共同研究の一部として、スイス放射性廃棄物管理協同組合が実施した共同研究成果に関するものである。

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** セーフティーアセスメントマネジメント

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1 INTRODUCTION

F.B. Neall

1.1 Background and aims

For those working within the field of radioactive waste disposal, safety assessments (SA) are a familiar tool for the evaluation of disposal concepts and potential repository sites. However, for those outside this group, whether in the scientific community or wider general public, there is often confusion about the aims, methods and results of safety assessment. In particular, there is a common perception that safety assessments seek to predict radiation doses over timescales of a million years or more. This is seen as fundamentally impossible, which undermines confidence in the implementing organisation and increases resistance to waste disposal programmes.

Therefore, the overall aim of this report is to present the H12 safety assessment in a way which makes the safety assessment process clearer and the implications of the results more meaningful both to workers within the SA field and to a wider technical audience.

This necessitates considering H12 and its results in a number of different ways to demonstrate:

- The reasonableness of the results:
  - Are the results comparable with results from other SAs of similar systems?
  - Are the results consistent with expectations arising from studies of natural systems?
- Confidence in the results of the assessment by showing that models and databases used are state of the art
  - Requires comparison with other recent SAs
- Redundancy within the natural and engineered barrier system (EBS):
  - Demonstration that safety is not dependent on the performance of a single part of the disposal system EBS.

The safety assessments used for comparisons addressing the first two aspects, above, include recent assessments from:

- Finland (TILA 99: Posiva, 1999)
- Sweden (SR 97: SKB, 1999a; SITE 94: SKI, 1996)
- Spain (ENRESA 97: Enresa, 1998)

as well as slightly older assessments from:

- Switzerland (WELLENBERG: Nagra, 1994a; KRISTALLIN-I: Nagra, 1994b)
- Belgium (UPDATING: Marivoet et al., 1992)
- Canada (AECL EIS; AECL 1994)
- Japan (H3: PNC, 1992).

Not all safety assessments are used in all comparisons; it has been found more useful to use specific SAs to illustrate particular points in the following chapters, rather than try to tabulate all the characteristics of all SAs. Thus, although this group of SAs includes assessments of different waste types (vitrified high level waste (HLW) and unreprocessed spent fuel (SF)), host rocks and disposal concepts, there is a common thread through all of them which can be used to illustrate safety assessment aims, strategies and results. Other recent SAs have been largely omitted from the comparison as, due to more fundamental differences in disposal concepts, their emphasis is somewhat different; this includes NIREX 97 (Nirex, 1997; cementitious repository for long-lived intermediate level waste (ILW)), Yucca Mountain Project (US DOE, 1999; unsaturated host rock) and Waste Isolation Pilot Plant (Sandia, 1996; ILW in bedded salt).

Throughout the text, the SAs are referred to only by name, e.g. H12, SR 97, TILA 99, Kristallin-I etc., without reference to the supporting documentation in order to avoid unwieldy repetition. However, the first section of the bibliography lists the main reports for all the SAs that are implied by the abbreviated references in the text. Where documentation other than the main report for a SA is referred to, this is given explicitly in the text. Also, for H12, which consists of an overview report and 3 additional supporting reports, references in the text note where a report other than the Overview is indicated.

As the results of the SA are given in terms of doses to a population over very long timescales, a further aim is to put both doses and timescales in a more meaningful context. Here it will be necessary to consider ways of demonstrating:

- The meaningfulness of calculations that give results for many thousands or even millions of years in the future
- To provide a framework timescale as a context for SA results over long times
- To demonstrate the smallness of the risk associated with the doses by comparison with other radiological (e.g. natural background radiation) and non-radiological risks. These non-radiological risks may be associated with everyday activities such as driving a car in Tokyo, travelling by Shinkansen (or "bullet train") or risks from earthquakes or volcanic activity while living in Japan.

In addition, it is considered appropriate and useful that perception of risk is discussed in this report. The overwhelming concern of a significant section of the population with small radiological risks is often considered "illogical" and, so far, the radioactive waste disposal industry has not really taken into account research which has tried to explain the relative importance of various voluntary and involuntary risks to the general public. However, this is a critical issue for public acceptance of radioactive waste disposal and must be considered when seeking to present safety assessment results "in perspective" to a wider audience.

Finally, throughout the report, the intention is to demonstrate how both natural analogue and experimental studies carried out by JNC for the waste disposal programme can be/have been integrated into the SA to support the results and conclusions of H12.

1.2 Structure of the report

An overview of the Japanese radioactive waste disposal programme is given in Chapter 2. This includes a discussion of the ethics of geological waste disposal (section 2.1) as well as a description of the context for the H12 study (section 2.2). A description of the H12 safety assessment follows (section 2.3). Additional sections discuss the evolution of models and databases from the H3 study to H12 (section 2.4) and the role of research and development (R&D) (section 2.5) in a waste disposal programme.

Chapters 3, 4 and 5 look at the H12 safety assessment in comparison with other assessments, considering the safety case, approach to SA and the results, respectively.

Chapter 6 is a discussion of timescales and the meaning of results calculated for long times. Chapter 7 begins with a division of risk perception with respect to radioactive waste disposal, which is followed by a section that attempts to show the significance of the doses calculated in H12 by comparison to other radiological hazards and also to everyday activities which carry risk of death or injury.

Finally conclusions drawn from the earlier chapters are summarised in Chapter 8.

It should be noted that the term "safety assessment" (SA) is used throughout this report in preference to mixing it with "performance assessment". This is for simplicity and consistency when writing for an audience somewhat wider than is usual for such reports. However, the terms are not strictly synonymous. According to the OECD[1] Nuclear Energy Agency (NEA) (NEA, 1997b):

performance assessment is defined as:

> "*Quantitative analysis of at least some subset of processes relevant to the behaviour of the disposal system and calculation of (at least) intermediate parameters of interest e.g. thermal evolution, container lifetime, contaminant release from some subpart of the disposal system. In addition, comparison of intermediate parameters to appropriate criteria set by regulation or design targets e.g. maximum allowable temperatures, minimum groundwater travel time, contaminant release from a subsystem.*"

and **safety assessment** is defined as:

> *"Quantitative analysis of a* ***set of processes that have been identified as most relevant to the overall performance of the disposal system*** *and calculation of a measure of overall performance relevant within the given national regulatory regime, e.g. individual dose to members of a critical group, integrated total release of contaminants. In addition,* ***testing of arguments that a sufficient subset of processes have been analysed, appropriate models and data used,*** *plus comparison of calculated measures of overall performance to regulatory limits and targets."*

1 Organisation for Economic Co-operation and Development

2 H12 - CONTEXT AND SUMMARY OF THE 2ND PROGRESS REPORT

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2.1 The ethics of geological disposal of radioactive waste

Radioactive wastes are produced, in one form or another, in every country in the world with significant industrial and medical facilities. Higher activity radioactive wastes are produced, in particular, from nuclear power generation and nuclear weapons programmes. Responsible management of all types of radioactive waste, in common with numerous other potentially hazardous industrial by-products, is one of the many environmental burdens which must be borne by society if it is to behave in an ethical manner towards the current and future health of the planet.

As with all other countries with nuclear power programmes, substantial volumes of the longer-lived and more highly active categories of radioactive wastes already exist in Japan, and will continue to be produced for many decades into the future, regardless of any decisions on the future use of nuclear power. It is the responsibility of Japanese society, in particular current generations which have benefited from nuclear power, to ensure that these wastes are managed safely and that undue burdens are not passed on to future generations. This is commonly known as the principle of "intergenerational equity". One way of ensuring that this principle is adhered to is to dispose of the wastes in an underground (geological) repository, in such a way that they can confidently be shown to pose no threat to the health of future generations or to the environment.

Such responsibilities are firmly incorporated in international principles for the safe management of radioactive wastes. For example, the International Atomic Enegy Agency (IAEA)'s Principles of Radioactive Waste Management (IAEA, 1995) state (Principle 5) that:

Radioactive waste shall be managed in such a way that will not impose undue burdens on future generations.

.....The responsibility of the present generation includes developing the technology, constructing and operating facilities, and providing a funding system, sufficient controls and plans for the management of radioactive waste.

......Limited actions, however, may be passed to succeeding generations, for example, the continuation of institutional control, if needed, over a disposal facility.

In 1994, the NEA addressed the issue of the environmental and ethical basis of the geological disposal of radioactive wastes in a special workshop (NEA, 1994), which led to the production of a collective opinion of their Radioactive Waste Management Committee (NEA, 1995a). Ethical aspects were considered by focusing on the issues of fairness and equity. The two principles which were developed at the time, and which have been much discussed since, can be defined as follows:

- *Intergenerational equity*: implying that it is the responsibility of current generations not to pass their problems and burdens on to future generations
- *Intragenerational equity*: meaning that, within current generations, it is important to ensure that our finite resources are spent sensibly on solving environmental problems, taking into account the relative scale of the potential impacts, and that decisions on

how to do this should be made in a fair and open manner, involving all sections of society.

Intergenerational equity is related to *sustainability*, or the sustainable development principle, which came into wide currency as a result of the UNCED[2] 1992 Rio de Janeiro consensus report (also called Agenda 21). Sustainability effectively means satisfying the needs of present generations without compromising the ability of future generations to meet their own needs. Agenda 21 essentially set the stage for much of the current direction of national and global environmental strategic thinking, and enshrined both sustainability and the *precautionary principle* firmly into international environmental policies. The precautionary principle implies that new activities that might degrade the environment should not be undertaken unless their impacts are well understood and acceptable, given the constraints of ensuring sustainability (i.e. we should "err on the side of caution").

The 1995 NEA report (NEA, 1995a) mentioned above considered that, with respect to long-lived radioactive wastes, responsibilities to future generations are better discharged by a strategy of final disposal than by reliance on permanent storage, which require surveillance, bequeath long-term responsibilities of care and may, in due course, be neglected by future societies whose structural stability can not be presumed. The NEA note that geological disposal is currently the most favoured disposal strategy.

Most recently, the NEA (NEA, 1999a) has discussed a further valid ethical concern (raised originally by KASAM[3] (1988), in Sweden) that goes hand-in-hand with the principle of intergenerational equity. This is that, whilst not passing on problems to future generations, we should equally not foreclose their options, or hinder their ability to make their own decisions. The consensus of the present community of technical experts is that a properly sited, designed, executed, sealed and decommissioned geological repository fulfils the objective of ensuring intergenerational equity. Effectively, it provides a sustainable solution that assures safety in a passive manner, requiring no further action or resource allocation from people in the future. However, it clearly also forecloses some options to future generations, which long-term storage would not. In fact, the technical feasibility of retrieval (even for a sealed repository) for times far into the future implies that no severe restriction on future options is imposed. On the other hand, long-term storage fails absolutely to meet the requirements of intergenerational equity. The apparent tension between these two concerns is compounded by some opponents of geological disposal, who believe that there are so many residual uncertainties surrounding its long-term impacts that it violates the precautionary principle. The argument is that society should not "rush into disposal" while scientific questions remain open.

There are two arguments which help to resolve this situation. First, the ethic of intragenerational equity suggests that present day resources should be used fairly. If an apparently good solution exists, then only limited effort is justified in improving it, considering competition for resources. The well-founded view of the expert scientific and technical community is that geological disposal is the optimum solution. However, this must be thoroughly explored and approved by the breadth of society, taking other, non-technical factors and other options into consideration, in order for it to be

2 United Nations Conference on the Environment and Development

3 Definition of KASAM needed

implemented. This is an area where resources are being spent today in many countries. Second, the way in which these ethical considerations are currently being interpreted is leading a trend towards progressive *stepwise* implementation of geological disposal, rather than any "rushing in". Each step of this process increases safety and reduces future burdens, but leaves critical decisions open to the generations that will exist at the time when specific actions are required.

Typically, this stepwise process might involve transfer of the waste to an underground repository, which is intrinsically safer than surface storage, but allowing flexible options during the period of time before the repository is closed, for example allowing for the easy retrieval of the waste. This allows future generations to become thoroughly comfortable with each decision that will be needed, at the time that it will be needed, based on the information that will then be available to them. It thus accepts the fact that, owing to the decades-long nature of disposal programmes, whatever the steps put in place now, they will inevitably be reconsidered, decided upon and implemented by other people.

An approach in which disposal is moved forward in a stepwise fashion, without abandoning further research, and maintaining reversibility as long as is required, can address all of the ethical concerns identified above. However, it requires very careful technical management over the full repository development programme, a well-organised and transparent system to ensure effective stakeholder involvement and a disposal concept that is robust yet flexible. The strategy should be inherently fail-safe at each step: adopting it must not compromise the basic safety concept upon which a repository programme is based, nor should it pass unallocated resource requirements to future generations to complete the process at the time they think fit. This is a fine balance.

The H12 concept for disposal of HLW can be seen as part of an ethically correct approach to managing Japan's most radioactive wastes. It aims at demonstrating that a solution is available: that geological disposal can ultimately be implemented in Japan and that undue burdens do not need to be shifted to the future population of the country or of specific regions. However, it is only an initial foundation stone for the stepwise programme which will eventually be required to achieve a widely endorsed solution to the disposal of these wastes.

2.2 Context of the H12 Report within Japan and with respect to other National Programmes

The organisational responsibilities for high-level radioactive waste management in Japan have been given to an implementing organisation (NUMO, Nuclear Waste Management Organization of Japan, formed in October 2000), whose responsibility will be to set in place and move forward a geological repository development programme. At present, NUMO focuses exclusively on vitrified HLW from reprocessing of spent nuclear fuel. There has, however, been a considerable amount of work during the last decade among the many interested and responsible government departments, committees, and other national institutions and nuclear industry organisations prior to the formation of NUMO. The H12 report presents a summary of progress by these groups up to 1999.

The basic policy for radioactive waste management, including guidelines for R&D activities, stems from the Atomic Energy Commission of Japan (AEC) and its "Long-term Program for Research, Development and Utilization of Nuclear Energy" (AEC, 1994). The Nuclear Safety Commission of Japan (NSC) is, on the other hand, responsible for guidelines for associated regulations. The Prime Minister is independently advised on policy development by the AEC and the NSC with their respective remits. At the time of production of H12, the Science and Technology Agency (STA)[4] acted as a secretariat for both Commissions.

JNC, the Japan Nuclear Cycle Development Institute (formerly PNC, Power Reactor and Nuclear Fuel Development Corporation) was charged by AEC with acting as the core organisation responsible for HLW geological disposal R&D (eg AEC, 1976 and 1993), and it has been carrying out research in this area for more than twenty years. It was not, however, responsible for either disposal of the wastes themselves (which predominantly belong to the power utilities) or for policy matters, which remained within the Government. As part of its R&D tasks, JNC prepared two progress reports on establishing a technical basis for HLW disposal in Japan: the so-called First and Second Progress Reports, designated as H3 and H12. These were produced in 1992 and 1999, respectively.

A sub-committee of the AEC (the Advisory Committee on Nuclear Fuel Cycle Backend Policy) set up the guidelines which JNC followed in producing the second, H12 progress report. These guidelines (AEC Guidelines) were published in 1997 (AEC, 1997). A previous AEC sub-committee (the Advisory Committee on Radioactive Waste Management) had set up the guidelines for the earlier, H3 study in 1989 (AEC, 1989).

In their 1997 guidelines, AEC set a number of objectives for JNC which were aimed at providing a baseline of information on a suitable disposal concept, with an analysis of overall performance and safety, and on factors in siting a repository. This information was intended to inform AEC and NSC as well as STA in their task of setting policy, technical and programme guidelines and regulatory guidelines for implementation of Japanese HLW disposal. It was also intended to inform the Ministry of International Trade and Industry (MITI)[5] in providing an implementation law for HLW geological disposal, which was promulgated in June 2000. This law prescribes the formal set-up of new implementing organisation (i.e. NUMO), how is it to be funded and what its responsibilities are. To co-ordinate work and pave the way for NUMO, the Government (STA and MITI), the utilities and JNC formed SHP (the Steering Committee for the High Level Waste Project) in 1995, which was phased out in July 2000. AEC noted that, given the urgency of the problem, it was essential to formulate concrete technical measures for geological disposal and to inform the public, clearly and transparently, of these measures with a view to obtaining their understanding and support. The role of H12 can thus be seen to be to support a somewhat complex decision making process which involved a number of influential parties, as well as the domestic broader public in Japan. In addition, the opportunity was taken to inform a wider, international audience by producing English versions of the 4 main reports[6].

4 After restructuring in 2000, both Commissions belong to the Cabinet Office

5 The MITI was restructured to the Ministry of the Economy, Trade and Industry (METI) in 2000. The STA became the Ministry of Education, Culture, Sports, Science and Technology (MEXT)

6 These were reviewed by an NEA expert group (NEA, 1999d and 1999e)

In parallel to JNC's work for H12, several other national R&D organisations and the utilities were carrying out closely related studies, and a Co-ordination Conference on R&D for Geological Disposal was formed in 1997 to help integrate the results. This acted as a steering group for the H12 project.

Simultaneously with setting up a HLW repository development programme, managed by NUMO, the Government needs to develop safety principles and a means of regulating the HLW programme. The NSC has started to discuss the regulatory safety principles and requirements, siting guidelines, safety criteria, technical standards and the licensing process and published its first report in 2000 (NSC, 2000). A further aspect of the H12 work was thus to make suggestions as to the nature and content of regulations for HLW.

H12 can be seen to lie at a relatively early and formative point within the Japanese national programme, and is intended to act as a foundation stone for developing policy and approaches, rather than being a statement of how the repository programme will necessarily evolve. Approximately equivalent parallel stages in other national HLW or spent fuel disposal programmes would be:

- KBS-3 (Sweden, 1983)
- Project Gewähr (Switzerland, 1985)
- AECL EIS (Canada, 1994)
- ENRESA 2000 (Spain, 1999)

Although there are differences of detail, each of these projects was carried out to demonstrate that a concept for disposal could be developed that would be appropriate for the specific wastes concerned and that repositories could be sited in the types of rock and geological environment found in those countries. They were all carried out prior to the initiation of site selection programmes and, in each case, it was noted that designs would need to be optimised and safety assessment methodologies developed as the programmes evolved. Like H12, several of these projects also built on earlier projects which had initiated the respective national programmes (e.g. KBS 1 and 2 in Sweden (KBS 1977 and 1978), VSE report in Switzerland (VSE, 1978)).

In this context, the guidelines set by AEC for H12 (Table 2.2.1) are a direct extension of the very general guidelines set for the earlier H3 study, as it was recognised in 1989 that at least ten years would be needed to carry out the necessary R&D.

Within the general requirements listed above, AEC identified numerous detailed issues and technical challenges which they wished to see addressed by the JNC studies. JNC interpreted the general guidelines, based on more specific comments in the 1997 AEC report, to arrive at the following objectives for the project:

1) To demonstrate that a suitable site for geological disposal of HLW can be found in Japan.
2) To demonstrate that the EBS and disposal facility as a whole can be constructed using currently available engineering technologies.
3) To demonstrate that the performance of the geological disposal system, with the emphasis on the near-field, can be reliably assessed.

Although the context and the objectives of H12 are thus broadly analogous to those of several national studies elsewhere, the Japanese programme has several unusual features and these need to be borne in mind when considering the work and comparing it with other studies:

- Japan is highly active tectonically, which makes future evolution of the geological environment potentially more difficult to evaluate than in some other countries and has a critical impact on approaches to selecting a repository site
- Japan will have considerably more waste to dispose of than many of the smaller national programmes (e.g. Switzerland, Belgium, Sweden, Finland, etc.), which may influence site selection due to the additional requirements for repository size
- H12 considers disposal of vitrified HLW only
- There is no regulatory framework yet to act as a guide for siting or safety studies.

2.3 Overview of H12

The H12 project comprises 3 major components, corresponding to the aims outlined above, namely a geological study and geosynthesis, an engineering design and feasibility study and a safety assessment. Each component was documented as a Supporting Report to the Project Overview Report. Aspects of all three areas are included in the following brief overview but, bearing in mind the aim of the present report, the emphasis is on the safety assessment and aspects of the other Supporting Reports which form input to it.

Tab. 2.2.1: The general guidelines set by the AEC for the H12 study (1997-2000) and the preceding H3 study (1989-1993)

H3 R&D Areas (AEC, 1989)	H12 Requirements (AEC, 1997)
Studies of the geological environment of Japan.	Specification of the characteristics of the geological environment important for disposal, and demonstration that suitable rock formations occur within Japan. Special emphasis on the stability of the geological environment and the near-field characteristics of the rock.
R&D on disposal technologies.	Specification of the design requirements for the EBS and other repository components and demonstration of the technical feasibility of meeting these requirements.
S+afety assessment studies.	Evaluation, with high reliability, of the performance of the system under the specific conditions of the geological environment of Japan.

2.3.1 HLW inventory for disposal

The baseline assumption of the Japanese fuel cycle is that spent nuclear fuel should be reprocessed to ensure the most efficient use of available resources. Spent fuel from the Japanese nuclear power plants is reprocessed in both foreign and domestic reprocessing plants. It is assumed that the vitrified HLW from reprocessing will be held in interim storage for 30 to 50 years, to allow cooling before disposal.

It is estimated that domestically reprocessed "JNFL waste" will represent the largest volume and, based on a comparison of thermal activity, neutron flux and hazard, it was considered reasonable to set the reference model inventory on the basis of the JNFL waste specifications. An analysis was carried out using the ORIGEN 2.1 code for conditions relating to the generation of JNFL waste. The inventory of vitrified waste to be disposed of by the year 2015[7] is predicted to correspond to approximately 40,000 packages, based on the nuclear power programme in Japan.

2.3.2 The geological disposal concept

The geological disposal concept forming the basis for the first progress report, H3, is basically carried over to H12. This consists of a multiple barrier system composed of a robust engineered barrier system (vitrified waste, overpack and buffer material) emplaced in a stable, geochemically favourable geological formation at a depth greater than several hundreds of metres (Fig. 2.3.1).

7 The MITI (now METI) modified its final disposal plan (MITI, 2000) so that the estimated inventory corresponding to 40,000 canisters of vitrified HLW is reached by the year 2020

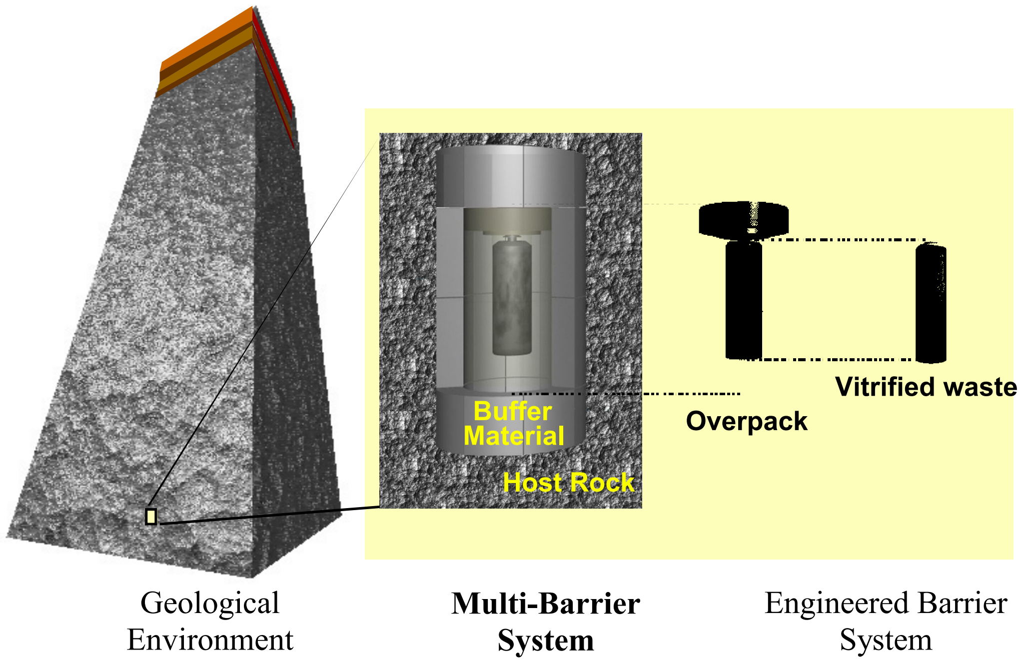


Fig. 2.3.1: Basic concept of the geological disposal system in Japan

As previously noted, no site or host rock type has yet been specified for the Japanese HLW repository and a wide range of rock types and geological environments were examined in the H12 study so as not to bias any later site selection process undertaken by the implementing organisation. As a consequence, the emphasis of the programme is on a highly reliable engineered barrier system (EBS). The possibility of testing the behaviour of the engineered barrier system by, for example, laboratory and in situ experiments, means that there is less uncertainty about its long-term performance than that of the geosphere. The geosphere is potentially a very effective barrier but there are uncertainties due to the inherent heterogeneity of geological structures, and its main role is to provide a favourable environment for the EBS.

2.3.3 Assessment timescales and safety indicators

With regard to timescales for safety assessment, the AEC Guidelines mention three time-related factors which have to be taken into consideration (AEC, 1997):

- Long-term changes in the human environment - The next glaciation is expected to occur around 10,000 years from now and will bring considerable changes to the biosphere and the near-surface environment. Although it may appear simplistic to assume current human lifestyles over periods in excess of 10,000 years, this is the only practical way to compare radiation doses in the far future with present-day doses.
- Long-term stability of the geological environment - The deep geological environment may be considered to remain stable beyond the next glaciation and it is therefore relevant to consider phenomena affecting the stability of the deep geosphere over periods of time extending far beyond the next glaciation.

- Characteristics of HLW - The spontaneous decrease in activity due to radioactive decay to some extent offsets the problems associated with the long timescales and future uncertainties involved in the safety assessment.

In H12, the primary safety indicator for the assessment is dose to a critical population without any time limit. Although setting the time limit for the safety assessment is normally the responsibility of the regulatory authority, a cut-off has been set at a point in time after the calculated dose reaches its maximum and the increasing uncertainties in the assessment at these long times are discussed.

For this discussion, the AEC Guidelines indicate that it is appropriate to include the results of supplementary analysis to confirm that there are no significant impacts on background radiation levels in the long-term, taking into account difficulties in predicting future human environment.

2.3.4 The Geological Environment in Japan

Based on the approach to research and development described in the AEC Guidelines, JNC identified the following major geological research targets for H12:

- To estimate future effects of natural phenomena that may degrade the deep geological environment, by considering past activity indicated by geological records
- To demonstrate that sufficiently stable geological environments for HLW disposal exist in Japan
- To summarise the general characteristics of rock formations and groundwater that are relevant to HLW disposal in Japan, using information collected in the fields of earth science and civil engineering, and from more detailed studies at JNC's Tono underground test site and their former site at Kamaishi.

Government policy states that a wide range of geological environments should be considered without specifying particular regions or rock types. However, within the context of HLW disposal, the geosphere is expected to provide:

- Physical isolation of the waste and, thus, a suitable site will rule out, as far as possible, failures of this isolation due to natural phenomena, e.g. uplift and erosion, or human intrusion
- A suitable environment (e.g. low groundwater flux, favourable groundwater chemistry, favourable thermal and mechanical properties) to allow the EBS to perform as intended and thus ensure long-term system performance
- A natural barrier to radionuclide release (low advective flow velocities, retardation, dispersion and dilution).

These requirements provide broad initial guidelines for examining the range of geological environments in Japan and identifying those which may or may not provide suitable repository sites.

2.3.4.1 The geological setting of Japan

Japan is located in the tectonically active circum-Pacific orogenic belt and, compared with stable regions such as those under consideration for repository siting in Europe, has generally high volcanic and seismic activity. This is due to the subduction and collision of tectonic plates, which cause active crustal movement. The geological environment of Japan is thus affected by gradual, large-scale phenomena, such as uplift, subsidence and denudation, and sudden, localised phenomena, such as volcanic and fault activity and earthquakes. This geological environment is also responsible for the rugged topography of Japan - about three quarters of the land area consists of mountainous regions (including hills and volcanoes). This situation should not, however, be considered as excluding siting: countries such as Switzerland also have relatively high seismicity, extensive uplift and mountain building due to such plate tectonics. The Yucca Mountain site in Nevada is located in an area clearly showing signs of recent volcanic activity.

Japan's rocks can be simply classified by distinguishing the older, pre-Neogene (more than 23 Ma old) basement complex from overlying Neogene/Quaternary sediments and volcanic rocks. Although generally less well studied, pre-Neogene rocks are cut by major tectonic lineaments which subdivide NE/SW Japan and the inner (continental) and outer (Pacific) sides of the country. Neogene and Quaternary rocks have been extensively studied as part of mineral exploration, water resource and civil engineering surveys. However, for the H12 safety assessment, crystalline rocks are divided into acid and basic, without consideration of age, while pre- and post-Neogene sediments are assigned a range of different physical properties and may be further subdivided into sandstone and mudstone/tuff categories[8].

2.3.4.2 Potential impact of geological phenomena

There are a number of natural phenomena that are particularly important to the feasibility of siting a HLW repository in Japan:

- Volcanism
- Fault movement
- Uplift (and associated denudation) or subsidence.

In addition, climate changes may also cause changes to the geological environment.

Localised phenomena such as volcanoes may be taken into account by use of a respect distance. The r*espect distance,* as used here, is the minimum distance from a volcano that a repository could be sited taking account of the requirement for long-term safety and uncertainty about the exact position of future activity and the unfavourable effects associated with them (e.g. high heat flow, dyke intrusion etc.). Respect distances may also be defined for other structures at a variety of scales. For example, for major fault zones (respect distances on the order of kilometres) due to uncertainty about future fault development and activity, or for sub-surface fractures (tens of metres) due to unfavourable local characteristics such as enhanced water flow.

These geological phenomena have been the subject of considerable effort to determine the patterns of past activity as a key to possible future impact. This has led to an understanding of the influence of these phenomena and associated processes, allowing an indication of how the key factors may be taken into account when siting a repository:

8 Based on JNC's own data (H12, Supporting Report 1), there appear to be ranges in key properties within each rock classification which are as great as the differences between them (e.g. Figs 3.2-6 / 3.3-5 from Supporting Report 1). There could be a good argument for a simpler classification into crystalline, consolidated sediments and unconsolidated sediments (c.f. Table 3.5-1, Supporting Report 1)

- It should be possible to avoid significant volcanic effects, which can probably be generally ignored at distances greater than a few 10's of kilometres from a volcanic centre. International consensus is that respect distance for volcanoes could be as low as 5-10km but the larger distance was used conservatively in H12. More realistic respect distances can be defined on a site-specific basis, from an understanding of the rate and direction of movement of the volcanic front[9]
- The associated effects of major fault movement on a deep geological repository are likely to be insignificant at distances greater than a few kilometres. As with volcanism, consensus is that a respect distance of around 1 km is sufficient but a larger distance was used in H12 with the reasoning that this could be reduced based on data collected during a site characterisation programme
- Areas with uplift rates much less than 1 m per 1000 years can be found
- Areas with denudation rates much less than 1 m per 1000 years can also be found
- A drop in sea-level of around 100 m within the next 100 thousand years must be taken into account (with cyclic changes having a period around 100,000 years)
- Significant glaciation is likely to be confined to Hokkaido and the upper slopes of the mountains in Honshu.

These individual siting considerations are similar to those identified in other countries although this particular combination is quite specific to the Japanese HLW programme.

2.3.4.3 Characteristics of the geosphere

The most relevant characteristics of the geological environment contributing to the multibarrier concept are those that influence the performance of the EBS and the functioning of the geosphere as a natural barrier. For example, rock mechanical and thermal properties are important for the design and construction of the disposal facilities and the eventual performance of the engineered barriers. The chemistry and flow of groundwater influence overpack corrosion and the leaching and migration of nuclides. Solute transport properties are particularly important aspects of the geosphere that determine its capacity to function as a natural barrier.

The geological environment needs to be represented by a conceptual model incorporating realistic information on:

- Hydrogeology
- Hydrogeochemistry
- Transport pathways
- Structure, lithology and rock physical properties.

Reasonable conceptual representations of the relevant phenomena have been developed using information from new (since H3) data compilations in conjunction with results from detailed investigations at JNC's Tono test site and former Kamaishi site. This information was also used to derive numerical values for key parameters, including hydraulic gradients, as well as properties of groundwater and rock formations. It is recognised that the actual values of parameters and importance of underlying processes

9 Volcanoes are related to the plate margin where subduction of one plate beneath Japan is taking place. The volcanic front links the line of volcanoes closest to the plate margin and it related to the depth of the subducted plate. The position of the front changes with time due to changes in the plate margin, rate of subduction or tectonic evolution

will be specific to the eventual site chosen for HLW disposal in Japan. However, the conceptual models and associated numerical information provide a basis for a reasonable first description of the types of geological environments that are likely to occur in Japan.

One general observation is that the two test sites, especially Kamaishi, and the associated general database probably tend to give a bias towards properties of "wet" sites. However, it should be noted that available data for appropriate (deep) rocks are almost non-existent as most data, excluding those from Tono and Kamaishi, originate from civil engineering projects at much shallower depths. Thus there is necessarily significant dependence on the dataset from Tono and Kamaishi which, although conservative from the SA viewpoint, might be somewhat over-pessimistic since it may well be possible to select a repository site with significantly more favourable properties. However, the point is that, if safety can be demonstrated for a relatively unfavourable generic site, there is more flexibility in the types of sites which can be considered for an actual repository.

It should be noted that there is also some inconsistency between the rock properties assumed for PA and tacit assumptions for the engineering study (which assumes very dry rock for some aspects of the study).

2.3.4.4 Physical properties of the host rock

In the absence of a specific disposal site, a variety of rock types having a wide range of physical and thermo-mechanical properties were considered in order to evaluate a range of design concepts, for example, indurated sedimentary rock with a relatively high mechanical strength and significant fracture porosity, or relatively uncompacted sediments with a low mechanical strength, significant connected matrix porosity but low fracture porosity.

However, for detailed design studies, a civil engineering classification, mainly based on the unconfined compressive strength, was used to define a "Soft rock system" and a "Hard rock system":

- Hard rock system: acidic crystalline rock, basic crystalline rock, pre-Neogene arenaceous sediments (sandstone), pre-Neogene argillaceous / tuffaceous sediments (mudstone / tuff)
- Soft rock systems: Neogene arenaceous sediments, Neogene argillaceous / tuffaceous sediments (5 different sets of physical properties considered).

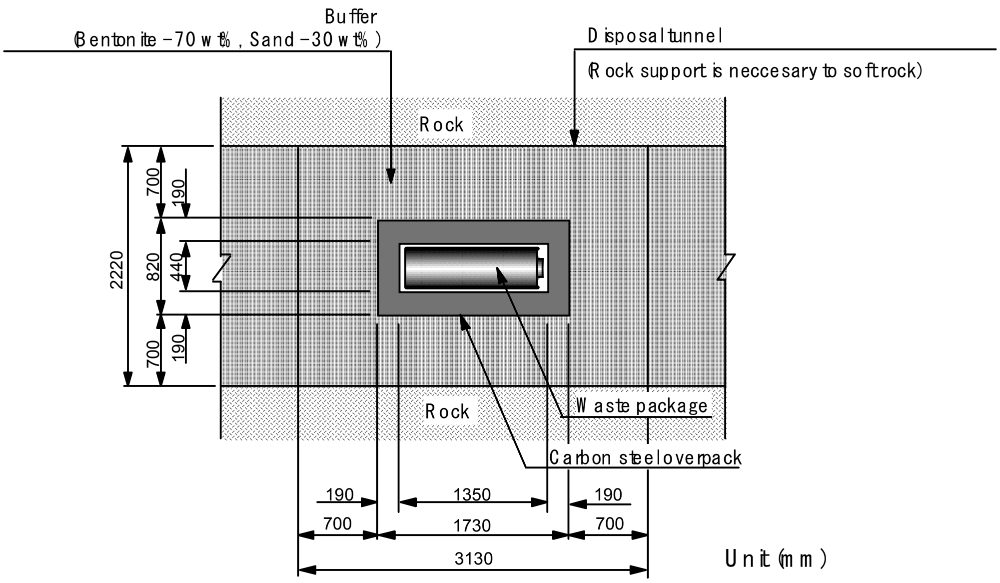
Two quite different datasets, for acidic crystalline rock and Neogene argillaceous/ tuffaceous sediments, were used to define reference "Hard rock" and "Soft rock" as a focus for engineering studies. Thus a large proportion of the rock types occurring in Japan were included between these two fairly extreme end-members used in the study.

It is very noticeable that the classification of different rocks in the H12 study focuses on lithology / age and mechanical properties rather than SA features, which are identified as critical in other SAs (in particular, hydrogeology). "Sandstone", for instance, can range from extremely high permeability rocks that form major aquifers to extremely low permeability rocks in which advective flow occurs only in fractures or other structural discontinuities. A consequence of this is that the link between SA and potentially favourable site characteristics has, to a large extend, been lost in the classification.

2.3.5 Reference HLW Repository Concept

The EBS was designed with a high performance margin to cover the wide range of geological conditions found in Japan. The EBS consists of the vitrified HLW, a rigid vessel (overpack) for containment of the vitrified waste and a buffer that fills the gaps between the overpack and the surrounding rock, as shown in Figure 2.3.2. It should be noted that both "in-hole" disposal (as indicated in Fig. 2.3.1) and "in-tunnel" disposal (as in Fig. 2.3.2, below) are considered in H12 and, although there is no difference in the dimensions of the EBS (hole and tunnel diameters, thus bentonite thickness, are the same), these disposal options behave somewhat differently with regard to safety[10]. It should also be noted that there are significant cost differences between the 2 options but these were not considered as optimisation of the EBS was not an issue in H12.

A carbon steel overpack is used for most design studies and analyses but alternative materials (Ti-steel and Cu-steel composite overpacks) are also considered.



Tunnel-disposal type (Hard rock / Soft rock)

Fig. 2.3.2: The engineered barrier system concept shown for the in-tunnel disposal option. Waste package[11] dimensions and bentonite thickness are the same for the in-hole disposal option

Four different waste package emplacement options were considered:

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10 This is due to the role of the excavation disturbed zone (EDZ) as a pathway for radionuclides released from the bentonite. In the case of "in hole" disposal, the disposal hole is excavated within the EDZ of the large diameter access tunnel from which disposal is carried out. Thus the host rock adjacent to these disposal holes is likely to be more extensively affected than that around the single small diameter tunnels used for "in tunnel" disposal. In H12, it is assumed that "in hole disposal" is intended to place waste packages beyond the influence of the EDZ around the large tunnel (see p.V-14 in Project Overview Report)

11 The combination of the HLW glass in its fabrication container enclosed in an overpack is commonly referred to as a "waste package"

1) Horizontal in-tunnel emplacement

2) Vertical shaft emplacement (2 or more waste packages "stacked" vertically)

3) Disposal hole horizontal emplacement (single waste package per hole)

4) Disposal hole vertical emplacement (single waste package per hole).

The in-tunnel horizontal emplacement option (i) minimises the excavation volume and the amount of backfill material required. On the other hand, the disposal hole vertical emplacement option (iv) is claimed to offer better operability in that emplacement of the waste packages and the buffer material is easier (although this would seem questionable based on recent experience from the Äspö underground laboratory in Sweden, SKB, 2002). Emplacement options can be selected flexibly or can even be combined, depending on the geological conditions at the site.

The in-tunnel horizontal emplacement method and the disposal hole vertical emplacement method were considered to provide a good contrast with one another and were selected as the reference options. In the SA, these options were considered to be equivalent, but this assumption is based on a lack of consideration of either the emplacement hole plug or the excavation disturbed zone (EDZ) of the large tunnel for the in-hole option.

The dimensions of the various EBS components were established by an iterative analysis procedure. Key constraints were:

- The waste form heat output (from waste loading of Japan Nuclear Fuel Ltd. (JNFL) glass and assumed 50 year cooling time before disposal)
- The presence of void space in the fabrication canister and overpack (from operational procedures)
- A desired overpack lifetime of at least one thousand years (to avoid requirements to quantify nuclide transport in the presence of significant thermal and water content gradients)
- A desire to keep the bentonite at temperatures below 100°C (to avoid possible loss of swelling properties)
- Specified rock thermal and mechanical properties.

The substitution of pure bentonite, used in H3, with a 70 % bentonite / 30 % sand mixture improves some important properties such as thermal conductivity and rheology (giving less risk of waste package sinking or bentonite erosion), while retaining the critical roles for the buffer of filtering colloids, ensuring solute transport by diffusion only, chemical buffering and sorption of key radionuclides.

Calculations indicated that a total overpack thickness of 19 cm of carbon steel was sufficient to provide the required radiation shielding and more than sufficient to withstand the expected pressure at a depth of 1000 m in hard rock or 500 m in soft rock, even assuming significant stress anisotropy, pressure from swelling of bentonite and overpack corrosion products, and allowing for 1000 years of corrosion in saline or fresh groundwater.

For the established EBS design, rock mechanical and thermal calculations were used to set the pitch of the emplacement tunnels and the waste packages, respectively. The stability of the EBS during seismic events was checked by analysis and large-scale experiments. The evolution of EBS properties during the early thermal period when groundwater saturation occurs were also assessed via coupled thermo-hydro-mechanical analysis.

It is noticeable that the EBS design does not evaluate implementation practicality – which could be problematic in either soft or hard rocks. Nevertheless, reference is made in the H12 report (H12 Supporting Report 2) to published EBS optimisation studies that offer options which may be more practical.

2.3.6 Assessing long-term safety

It is important to recognise that the main aim of the H12 safety assessment was to demonstrate the assessment technology through application to generic geological conditions which are representative of the type of locations which might be selected as repository sites (although, as noted in section 2.3.4.3, concentration on Tono and Kamaishi is probably very conservative). As output, the analysis indicated not only the fundamental feasibility of safe disposal but also both the performance criteria that should be used to guide site selection and the priority areas for associated R&D.

The safety concept examined in this study emphasised the performance of the repository near field, particularly the engineered barrier system, reflecting the generic nature of the host rock properties.

2.3.6.1 Development and treatment of safety assessment cases

The H12 safety assessment considered a range of disposal systems, including different possible geological and surface environments and repository designs. Furthermore, the effects of various sources of uncertainty on system performance were considered. This leads to a potentially very large number of cases if all combinations of geology, surface environment and repository design are considered. Thus, as in SAs carried out in other HLW disposal programmes, it was necessary to establish a Base Scenario and Reference Case, around which sensitivity analyses were performed deterministically, in order to reduce the cases considered to a realistic number.

The assessment consisted of the following steps:

- Development and application of a systematic methodology to ensure that the relevant features, events and processes[12] (FEPs) were fully taken into account in developing scenarios for the assessment
- Definition of a Base Scenario (one of a number of groundwater scenarios; see Fig. 2.3.3) and, within this scenario, definition and quantitative analysis of a Reference Case and alternative cases

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12 Features, events and processes are the building blocks of scenarios. Hence the Base Scenario, in which radionuclides are released from the EBS and transported by groundwater through the geosphere, involves a large number of FEPs including the description of the EBS and geosphere ("features" of the system) and "processes" such as overpack corrosion, waste matrix dissolution and transport of the dissolved radionuclides through both the bentonite and the rocks of the geosphere. "Events" includes both naturally-occurring phenomena, such as earthquakes, and also those such as human intrusion which might contributre to failure of isolation. "Transport" includes the physical processes of diffusion and advection as well as chemical processes such as sorption, precipitation and dissolution of nuclides

- Definition of a range of isolation failure scenarios[13] and alternative ("perturbation"[14]) groundwater scenarios for either quantitative or qualitative analysis
- Indication of key phenomena and uncertainties from the results of these analyses
- Overall assessment of the system performance in a range of geological and surface environments, particularly with respect to the feasibility of safe geological disposal in Japan.

13 Isolation failure scenarios are those which consider events and processes which could lead to breaching of the repository. This could be due to processes of uplift and erosion, or an event such as human intrusion by, for example, bore hole drilling

14 Perturbations are deviations from the normal or expected evolution of the repository system. This might involve, e.g. failure of seals leading to alternative groundwater flow pathways through the repository, or climate changes leading to changes in hydrological conditions around the repository

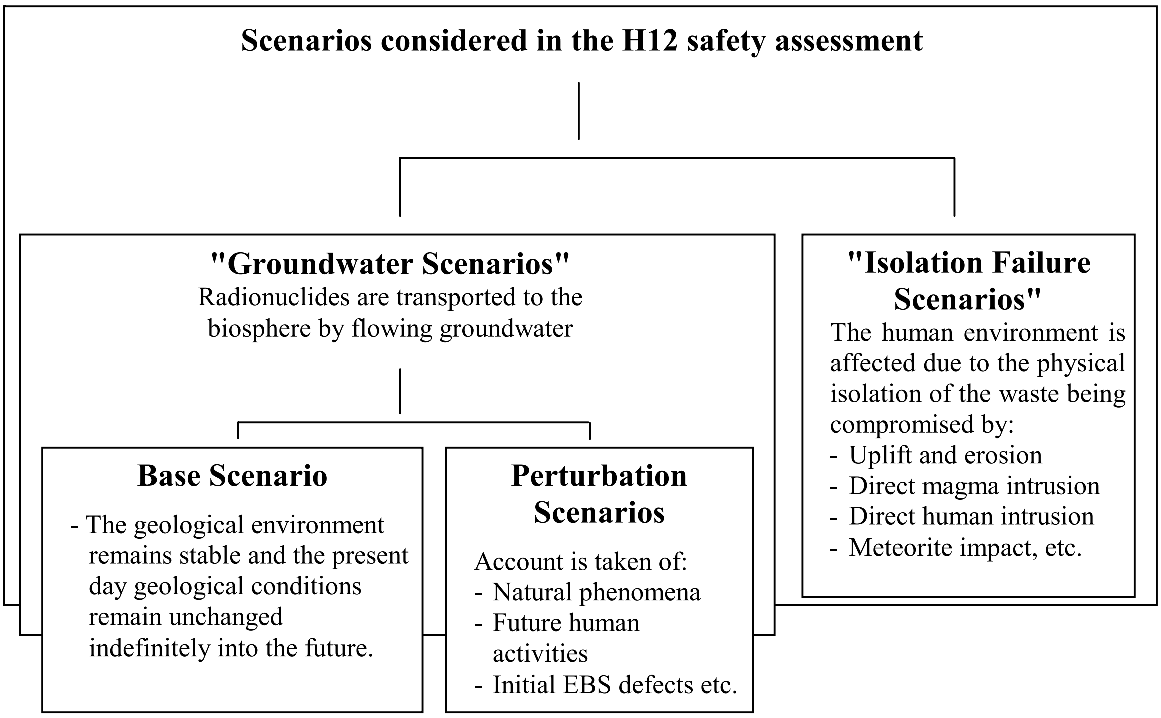


Fig. 2.3.3: Classification of scenarios

The following calculation cases were considered within the Base Scenario:

- The Reference Case
- Alternative geological environments and alternative design cases to address various geological disposal systems
- Model and data uncertainty cases.

Model and data uncertainty cases were selected based on an evaluation of their potential significance. In addition, some of the model uncertainty cases include FEPs that could be relevant, but which are not considered in the Reference Case.

In addition, a number of calculation cases are considered for the perturbation scenarios. Isolation failure scenarios are treated either qualitatively or by less formal "what if?" calculations. Because of the generic nature of this study, clear emphasis was on the evaluation of the consequences of perturbation / isolation failure scenarios rather than assessing their probability.

2.3.6.2 Modelling strategy

The near field, consisting of the EBS and a limited volume of the surrounding host rock, is the element of a repository system that tend to be characterised by greatest certainty. In H12, this led to the development and application of relatively realistic near field datasets and models, although with moderately conservative assumptions made where there was uncertainty. In the case of the surrounding geosphere, greater uncertainties associated, for example, with characterisation of large scale heterogeneity from a limited number of in-situ measurements, led to a more conservative modelling strategy. Emphasis on the near field was considered appropriate given the lack of relevant data for deep rock formations, prior to site selection.

A different approach was taken for the biosphere assessment. No attempt was made to model the evolution of the surface environment and the lifestyles of future generations, due to uncertainties that are largely irreducible. Rather, certain sets of assumptions are made on these aspects of biosphere modelling, giving rise to stylised representations of the biosphere. These were used to convert radionuclide fluxes to doses and thus provide a means to evaluate the radiological consequences of geological disposal.

Figure 2.3.4 illustrates the chain of models used directly in the safety assessment or to produce key data, and the associated flow of information between models. This model chain was used for the Reference Case calculations and for calculations of many of the alternative cases.

2.3.6.3 The Reference Case

The Reference Case was based on the Base Scenario and the reference design outlined in Section 2.3.5 and summarised in Table 2.3.1. Prior to assigning parameter values, a database was developed by collating relevant experimental measurements and observations. The values used for the Reference Case calculations were selected from ranges within the database by experts in the relevant scientific fields.[15]

15 Much of the data / database selection procedure was subject to independent review by a Nagra team during production of H12. A formal comprehensive Quality Assurance (QA) procedure was, however, in the development stage and hence data review / traceability is not fully guaranteed

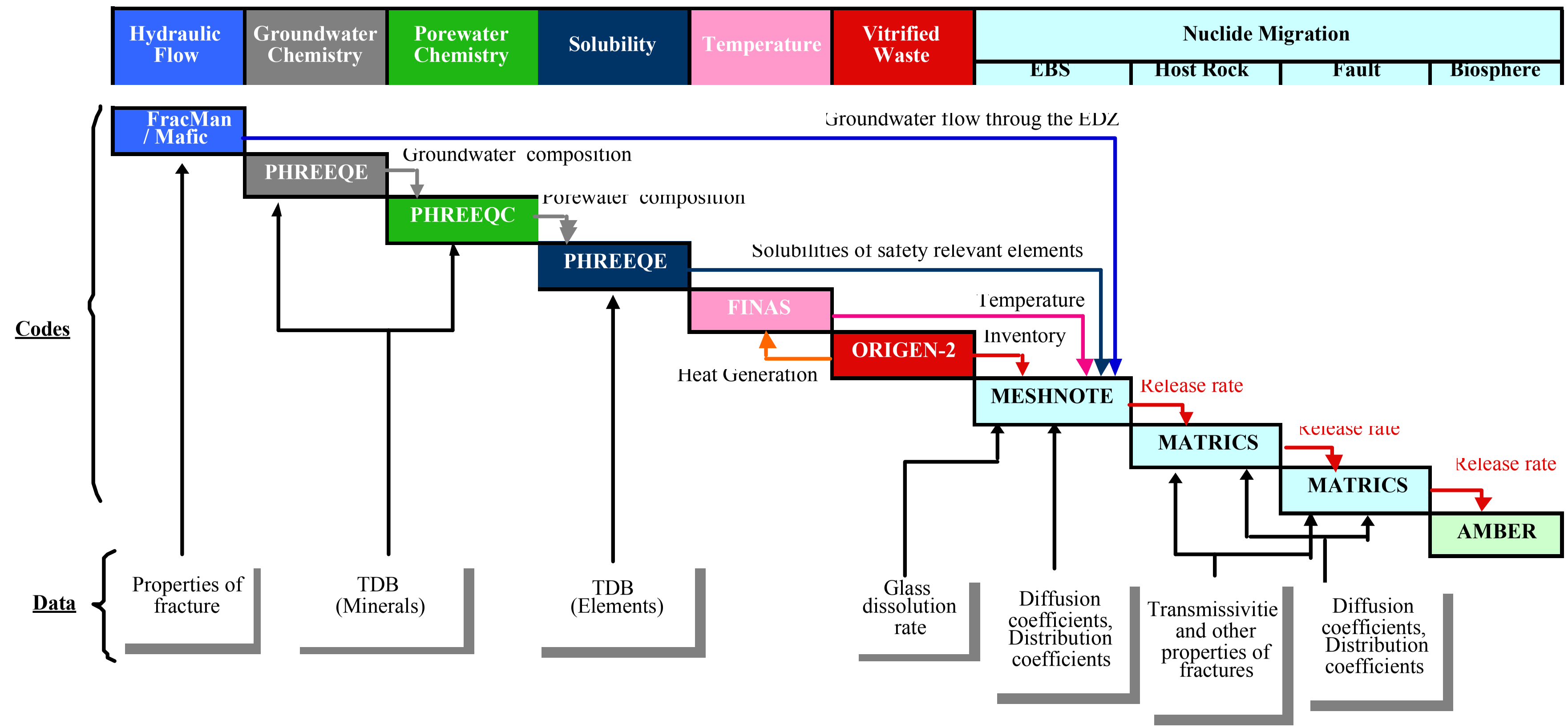


Fig. 2.3.4: Model chain used in the H12 safety assessment, and associated flow of information (H12 Supporting Report 3) (TDB: Thermodynamic database)

Tab. 2.3.1: System components considered in the safety assessment

Geological / Surface Environment	System components considered in safety assessment	Reference set of system components (Reference System)
Topography	➢ Mountains ➢ Hills ➢ Plains (plateau, lowlands)	Plains (lowland)
Groundwater	➢ Fresh groundwater ➢ Saline groundwater	Fresh groundwater
Rock type	➢ Crystalline: felsic, mafic ➢ Sedimentary (soft rock): sandstone, mudstone/tuff ➢ Sedimentary (hard rock): sandstone, mudstone/tuff	Crystalline rock (felsic)
Geosphere-biosphere interface	➢ Surface water (river, lake etc.) ➢ Well ➢ Marine	Surface water (river)
Design	**System components considered in safety assessment**	**Reference System**
Vitrified waste	Model vitrified waste based on JNFL specifications (assuming 40,000 containers)	
Overpack	➢ Carbon steel ➢ Composite: titanium – carbon steel, copper-carbon steel	Carbon steel: 0.19 m thick (with minimum design lifetime of 1000 years)
Buffer material	➢ Mixture of bentonite and silica sand	Bentonite 70 wt% + silica sand 30 wt%, dry density 1.6 Mg m^{-3}, thickness 0.7 m
Repository depth	➢ Up to approx. 1000 m for hard rock ➢ Up to approx. 500 m for soft rock	1000 m for hard rock
Temperature at disposal site	➢ The temperature will reach about 55ºC after 1000 years in the case of hard rock, and about 40 – 50ºC after 1000 years in soft rock.	After 1000 years, constant at 60ºC in the EBS and 45ºC in the geosphere
Others	➢ Tunnel supports: concrete, steel ➢ Drift excavation technology: blasting, mechanical drilling (tunnel boring machine) ➢ Backfill material: mixture of bentonite and crushed rock with controlled particle size ➢ Plugging material: compacted clay, cement, rock, metal ➢ Grouting material: clay, cement	It is assumed that tunnels do not require supports in a hard rock environment

EBS Reference Case

The Reference Case conceptual model assumptions for radionuclide transport in the EBS were:

- The overpack fails 1000 years after closure of the repository
- The glass waste form makes contact with water immediately after overpack failure. Nuclides in the glass are released congruently as the glass dissolves at a constant rate. The decrease in the glass surface area due to its volume decrease is ignored
- Aqueous concentrations of nuclides are limited by elemental solubility, shared among the isotopes contained in the glass. Stable isotopes in the groundwater / bentonite porewater are not considered. Dissolution and precipitation of nuclides occur much faster than their (diffusive) transport and achieve local equilibrium
- Nuclides are transported through the buffer material by diffusion and are retarded by linear, reversible and instantaneous sorption on the bentonite. The diffusion resistance of the overpack corrosion products and their sorption capacity are ignored
- Nuclides released from the buffer are mixed instantaneously and completely with groundwater flowing through the excavation disturbed zone (EDZ) of the repository tunnels. Sorption of nuclides in the EDZ is not considered and thus the total nuclide flux enters fractures intersecting the EDZ.

A summary of EBS Reference Case data is given in Table 2.3.2; Figure 2.3.5 shows the release rate from the EBS for a single waste package in the Reference Case.

The high solubility (assumed unlimited in this analysis) and small distribution coefficient of Cs-135 means that it dominates release rates between one thousand and five hundred thousand years after repository closure. Nb-93m, which is assumed to be transported in radioactive equilibrium with its parent Zr-93, becomes dominant after 0.5 Ma.

Isotopes with large elemental inventories and low solubilities, e.g. Np-237 and Tc-99, precipitate in the vicinity of the glass. This results in a low and pseudo-steady state release for a prolonged period as the concentration of these isotopes in the bentonite porewater will be controlled by their elemental solubility limit. Also dissolution of the accumulated precipitate may last for a significant period after complete dissolution of the waste glass. Relatively short-lived isotopes (half life shorter than several tens of thousands of years) of highly sorptive elements, e.g. Pu-240 and Am-241, decay significantly within the buffer material and their peak releases are extremely small.

Tab. 2.3.2: Summary of data used in the EBS Reference Case model

Process / Feature	Parameter	Data in the Reference Case
Glass dissolution	Glass dissolution rate	1 x 10^{-3} g m^{-2} per day
	Glass surface area	17 m^2 per WP (10 times the original geometric surface area of glass)
Migration in the buffer material	Elemental solubilities	See discussion in section 4.5.4
	Elemental distribution coefficients	See discussion in section 4.5.4
	Effective diffusion coefficients	Se: 2 x 10^{-10} m^2s^{-1} Cs: 6 x 10^{-10} m^2s^{-1} Other elements: 3 x 10^{-10} m^2s^{-1}
Release to the surrounding host rock	Groundwater flow rate in EDZ	1 x 10^{-3} m^3 a^{-1} per waste package
Engineered barrier system design	Inner radius of buffer material	0.41 m
	Outer radius of buffer material	1.11 m
	Waste package pitch (in-tunnel disposal)[1]	3.13 m
	Porosity of buffer material	0.41
	Dry density	1.6 Mg m^{-3}

Note:
1: Waste package (WP) pitch is the distance between the midpoints of adjacent waste packages for in-tunnel disposal and is equivalent to a 0.7 m thickness of bentonite at the end of each waste package (WP length (1.73 m) + 2 x 0.7 m). The length of buffer material for in-hole disposal will be 3.13m plus a further thickness of buffer at the top of the hole to give additional radiation protection during operations in the tunnel.

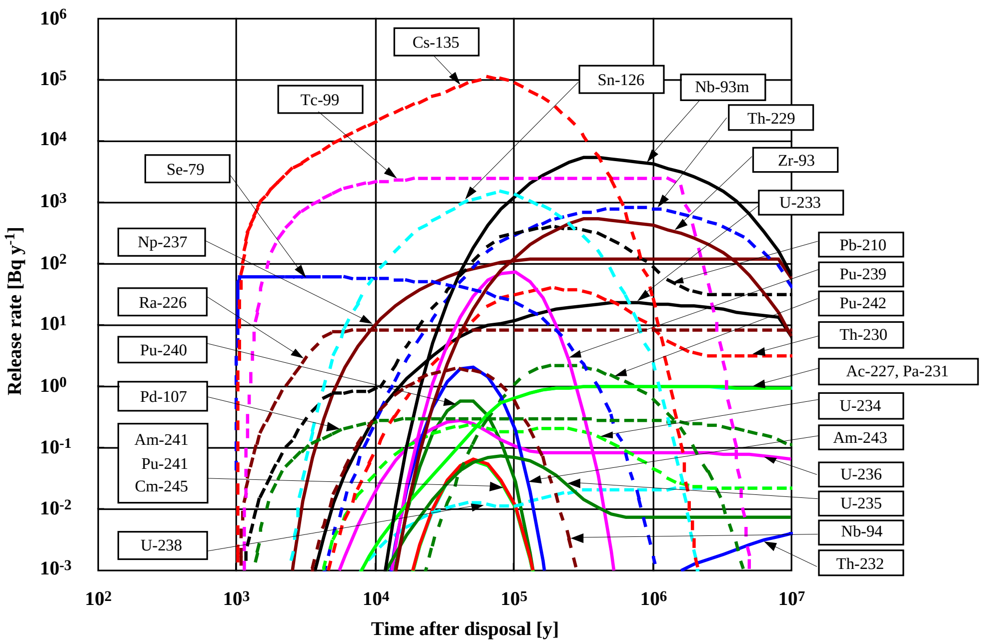


Fig. 2.3.5: Calculated release rate in Bq per year from the EBS for a single waste package in the Reference Case

Geosphere Reference Case

The conceptual model for nuclide transport through the Reference Case host rock (granite) was developed based on the following assumptions regarding transport-relevant processes:

- Nuclides are transported as solutes through channels within a network of fractures in the rock
- Hydrodynamic dispersion due to the complex network structure of the fractures is taken into account through a macroscopic dispersion length
- Nuclides transported in channels can diffuse into the adjacent rock matrix. Nuclides may also be retarded by sorption on matrix pore surfaces. Sorption is assumed to be linear, reversible and instantaneous. Sorption on fracture surfaces or fracture infill is not considered
- Diffusion into fracture filling materials and unconnected fractures is not included.

The geosphere is modelled assuming a thickness of undisturbed[16] host rock around the repository with an outer region in which the host rock is cut by major water-conducting faults. The transport properties of the faulted region (i.e. the transport properties of the faults) may be significantly less favourable than in the undisturbed host rock where transport is assumed to be dominated by relatively poorly connected channels in much smaller fractures. Radionuclides were assumed to be transported initially through these channels in undisturbed host rock before reaching a single major water-conducting fault, along which they are transported by a similar set of processes, upwards to an aquifer. In the Reference Case, the flow path length in the undisturbed host rock is taken as 100 m and in the fault, 800 m.

A summary of the geosphere Reference Case data is given in Table 2.3.3.

Figure 2.3.6 shows the calculated release rates for nuclides from the host rock to the fault (2.3.6 (a)) and from the fault to the biosphere (2.3.6 (b)) in Bq per year for a single waste package in the Reference Case. In both cases, Se-79 dominates for a short initial period before being overtaken by Cs-135. Both nuclides have high release rates from the EBS and low sorption in the geosphere; Cs-135 also has a relatively long half life. It should be noted that the peak releases for these radionuclides (as well as others) are delayed by transport through the fault but that the magnitude of the peak releases are almost unaffected, due to their low sorption and long half lives.

16 "Undisturbed" host rock here implies host rock outside the excavation disturbed zone (EDZ). However, it still includes small fractures and water-conducting pathways in which transport can take place

Tab. 2.3.3: Summary of Reference Case geosphere data

		Undisturbed host rock	Major water-conducting fault
Geological environment	Rock type	Granite (felsic crystalline rock)	
	Groundwater	Fresh type groundwater	
	Hydraulic gradient	0.01	
Migration path		through the host rock from the edge of the repository to a downstream fault	migration along the fault from the repository level to the biosphere
Nuclide source		release from the EBS	release from the host rock
Data	Migration length	100 m	800 m
	Transmissivity	$10^{-13} \sim 10^{-7}$ $m^2 s^{-1}$ (Velocity = 0.05~ 50 m a^{-1})[1]	10^{-7} $m^2 s^{-1}$ (velocity = 50 m a^{-1})
	Longitudinal dispersion length	10 m (migration length x 0.1)	80 m (migration length x 0.1)
	Proportion of fracture surface area available for matrix diffusion	0.5	
	Matrix diffusion depth	0.1 m	
	Porosity [2]	0.02	
	Dry density [2]	2.64 Mg m-3	
	Effective diffusion coefficient [2]	3 x 10–12 m2 s-1	
	Distribution coefficients	See discussion in section 4.6.3	

Notes:

1: This velocity is given assuming parallel plate fractures, and calculated using the equation:

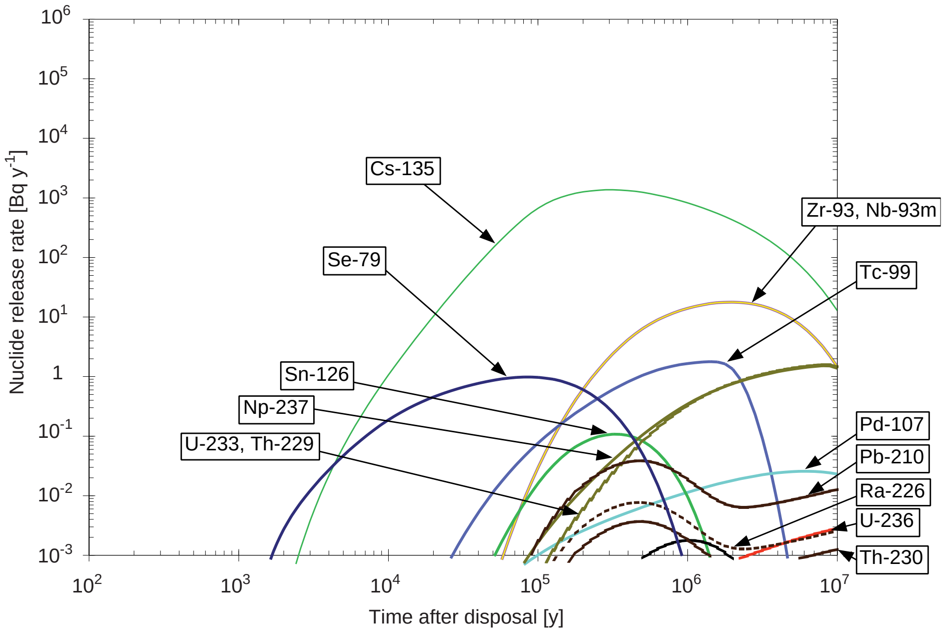
$$\text{velocity } (m\ s^{-1}) = \text{transmissivity } T\ (m^2\ s^{-1}) \times \frac{\text{hydraulic gradient}}{\text{fracture aperture (m)}}$$

The fracture aperture is defined by the empirical law as $2T^{1/2}$

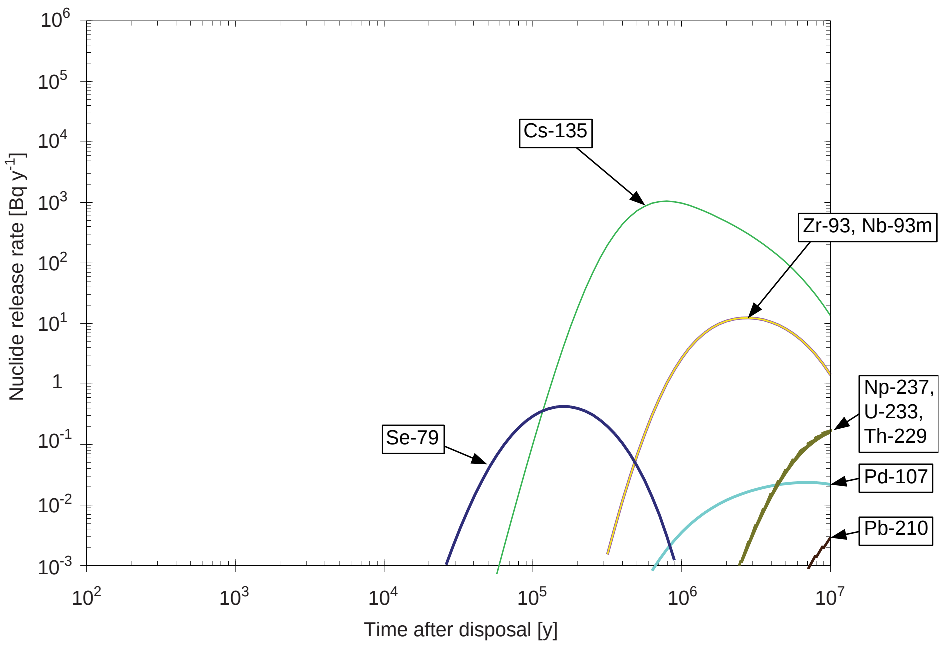
2: The relationship between porosity, dry density and effective diffusion coefficient is taken into account.

Biosphere Reference Case

In the Reference Case, nuclides that have been transported through the undisturbed host rock and then a major water-conducting fault enter an aquifer near the ground surface and from there are released to the biosphere via the geosphere-biosphere interface (GBI). A river in a plain (lowlands) was selected as the Reference Case GBI, since around 80% of the Japanese population is concentrated in such areas and rivers represent the dominant water source for both human consumption and agriculture. Assumptions concerning the surface environment and human lifestyle in the future are summarised in Table 2.3.4.



(a) Calculated release rate from the host rock to the fault per waste package



(b) Calculated release rate from the fault to the biosphere per waste package

Fig. 2.3.6: Calculated nuclide release rate from the geosphere per waste package (Reference Case)

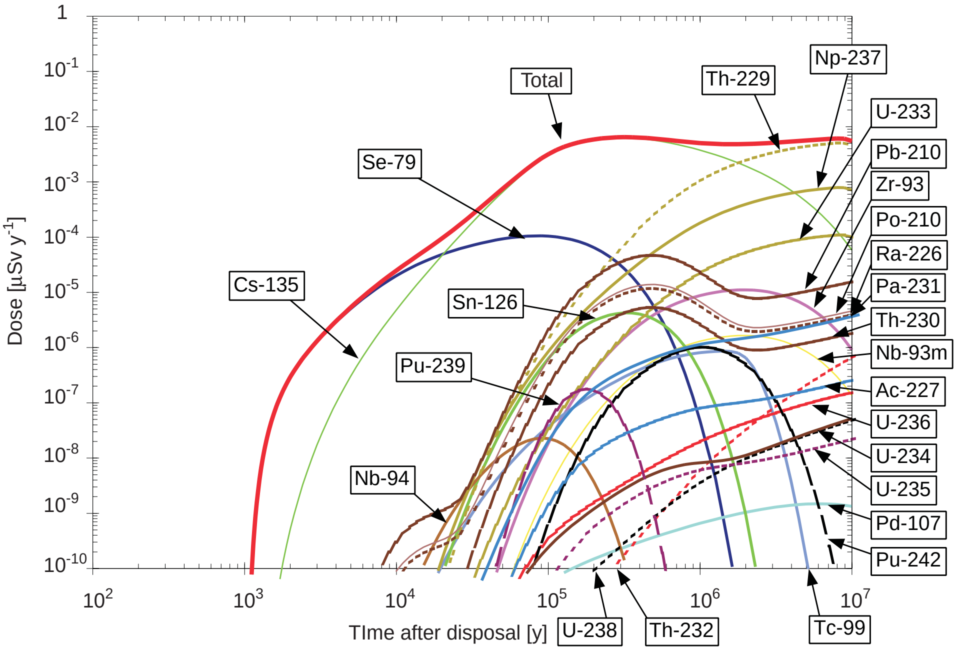
Tab. 2.3.4: Assumptions concerning the surface environment for the biosphere assessment in the Reference Case

Location	➢ Japan, but non-site specific
Geographical setting	➢ Farmland, irrigated by contaminated water abstracted from a river, as well as river and coastal marine environments downstream from the repository exfiltration zone
Topography	➢ Either a lowland plateau or a coastal environment, where the land is gently sloping and covered with soil
Climate	➢ Present day Japanese temperate climate (no climate change considered)
Geomorphological processes	➢ Those occurring in the relevant environment in present day Japan
Land use	➢ Agriculture in the case of farmland, and fisheries in the case of river and coastal environments
Life style	➢ Present day Japanese societal conditions, with locally produced and consumed produce; ➢ Water quality tests are not conducted for either drinking water or water drawn from wells and used for irrigation

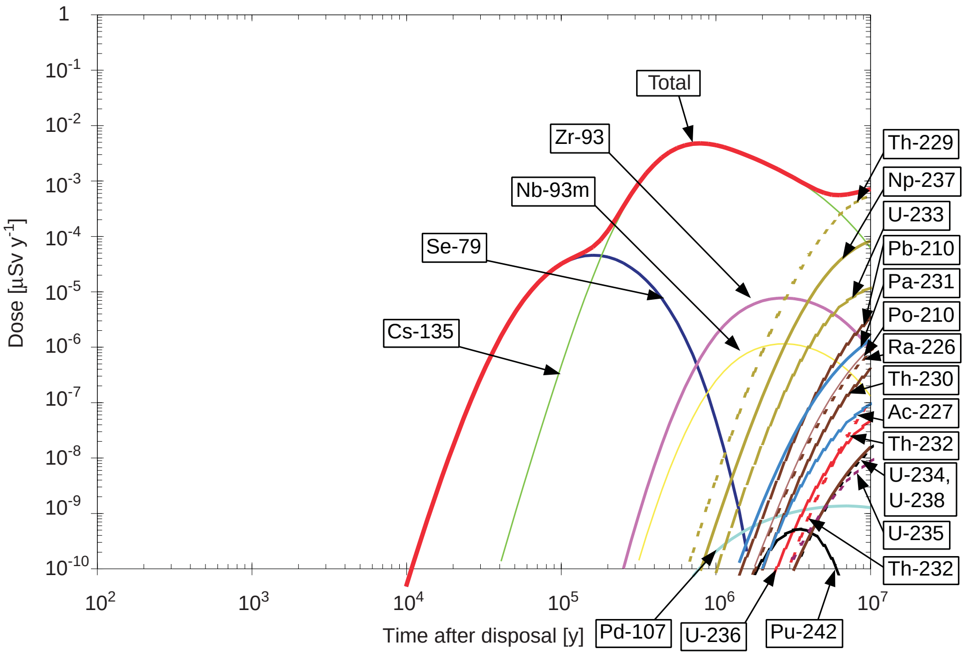
In calculating doses to humans, exposure pathways were first defined, based on the way in which radionuclides in river water are assumed to be distributed within various biosphere components. Characteristics were then assumed for the human population, which represent plausible human behaviour with consumption of locally produced foodstuffs. The Reference Case considered the following exposure groups:

- Farming exposure group: a group of subsistence farmers who live on local agricultural and dairy products
- Freshwater fishing exposure group: a group of fishermen along the river who subsist predominantly on their freshwater products
- Marine fishing exposure group: a group of fishermen who subsist predominantly on their marine products.

Figure 2.3.7 shows the estimated potential dose history for a repository of 40,000 waste packages in the Reference Case for the geosphere with and without the faulted host rock included in the modelling. In Figure 2.3.7(a), radionuclides are released from the undisturbed host rock directly to the biosphere, without transport in the fault. In contrast, Figure 2.3.7(b) shows the effect of including transport in the fault before radionuclides are released to the biosphere. The overall maximum calculated doses are 0.006 $\mu Sv\ a^{-1}$ and 0.005 $\mu Sv\ a^{-1}$, respectively, although the time to peak dose is much longer in the case with the fault. In each case, Se-79, Cs-135 and Th-229 (in equilibrium with Np-237) dominate the calculated dose in turn as time progresses, with the maximum dose being dominated by Cs-135.



(a) Calculated dose corresponding to release rate from the host rock to the fault



(b) Calculated dose corresponding to release rate from the fault to the biosphere

Fig. 2.3.7: Calculated dose corresponding to release rate for 40,000 waste packages. In (a), radionuclides are released directly from the undisturbed host rock to the biosphere, whereas in (b), radionuclides are transported through the major water-conducting fault to the biosphere (Reference Case; Po-210 is evaluated assuming radioactive equilibrium with Pb-210)

2.3.6.4 Alternative Cases

An extensive set of alternative calculation cases was analysed in order to study the sensitivity of the results to uncertainty in models and data for both EBS and geosphere, and to examine alternative disposal system designs and geological environments. The results of these alternative cases are too extensive to give here and the reader is directed to the H12 Overview and Safety Assessment (Supporting Report 3) reports. However, some points of interest are:

- Increasing or decreasing the overpack lifetime by an order of magnitude influences only the time of breakthrough of the most mobile radionuclides (Cs-135 and Se-79) but has negligible effects on the calculated dose maximum
- Increasing or decreasing the glass corrosion rate has an influence only on non-solubility-limited elements, for which the variation in dose is most significant for relatively poorly sorbed isotopes (i.e. for Cs-135 rather than Am-243)
- Due to the tendency of sorption to decrease (in both the near- and far-field) in more saline waters, the maximum release rate of Cs-135 for a saline system is a 3-4 times higher than the freshwater reference. However, the release rate is limited by the Cs leaching rate from the glass
- Maximum dose is very sensitive to both ground water velocity (or transmissivity) and factors affecting matrix diffusion such as matrix diffusion depth, area of fracture surface available for matrix diffusion and diffusion coefficient. Fracture aperture is an insensitive parameter since the effects of groundwater velocity and matrix diffusion efficiency counteract each other
- A critical geosphere/biosphere parameter is the contrast between water fluxes through the repository and the output aquifer (i.e. dilution). The calculated doses scale directly with this parameter.

2.4 Development / evolution of concepts, models and databases from H3 to H12

Concepts

The design requirements for the EBS and the general disposal facility were determined based on utilisation of currently available technology, taking economic aspects into consideration. Since the publication of H3, more reliable supporting data have been obtained from demonstration tests on both laboratory and engineering scales (carried out at JNC's ENTRY facility, etc.). Design requirements have been reviewed, the analytical design tools have been improved and the design database has been extended to provide a better understanding of the barrier functions of the EBS. Based on these refinements, it was considered reasonable to reduce the thickness of both the overpack and the buffer material by approximately 30% compared with the specifications in H3 (Fig. 2.4.1).

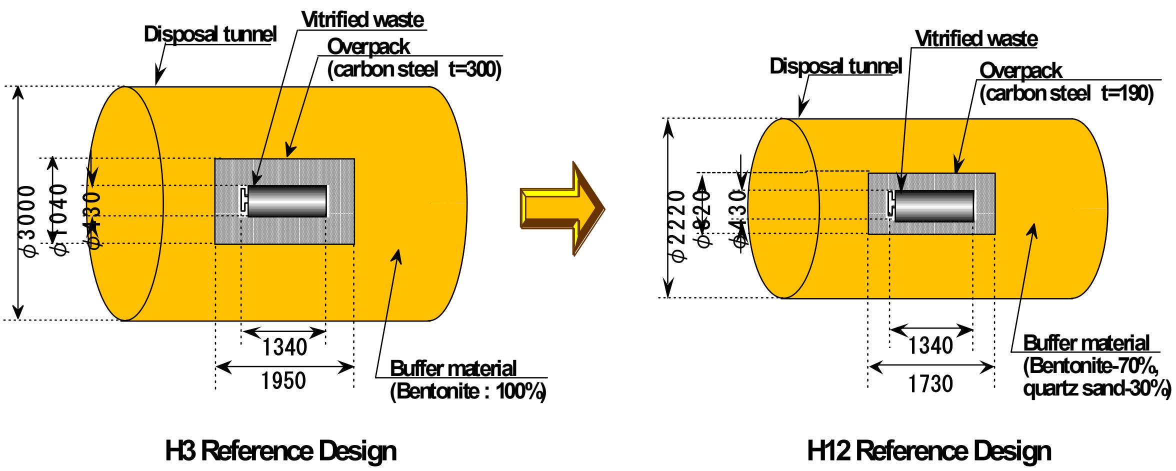


Fig. 2.4.1: Evolution of the EBS design from H3 to H12 (dimensions in millimetres) (From Umeki, 2000)

This leads to a reduction of around 50% in the total volume of EBS materials. Bentonite mixed with quartz sand was selected as the buffer material, bringing about a reduction in costs without compromising performance. It should be noted that such optimisation arguments were not used to distinguish between the various emplacement options (in particular, in-hole versus in-tunnel) which clearly have very different costs (the former are much more expensive).

Scenario development

In H3, SA calculations focused on a base case of groundwater release scenarios. However, these scenarios were only listed and described, without showing that a systematic procedure has been used to develop them.

In H12, in order to reduce the risk of overlooking potentially important scenarios, a systematic methodology was developed and applied. In this methodology, a comprehensive list of FEPs was first developed by collating the FEP lists developed in other projects (e.g. Nagra, 1994b; NEA, 1999c). Then, state-of-the-art scientific knowledge, a large body of scientific literature and JNC laboratory and in-situ experimental results were used to develop greater understanding of the safety functions of system components and potentially detrimental factors. Screening criteria were then applied to narrow the range of FEPs for inclusion in the consequence analysis, i.e. to exclude:

- FEPs that are unlikely to affect the safety of geological disposal provided an appropriate geological environment is selected
- FEPs that can be avoided by appropriate design and construction of a repository and by engineering measures
- FEPs whose probability of occurrence is extremely low.

The FEPs that are not screened out were used to define a number of scenarios that provide the basis for modelling system performance under certain well-defined assumptions (including the operation of safety functions and potentially detrimental factors).

The only scenarios modelled in detail in H12 were "groundwater scenarios"; not only a Base Scenario but also a set of perturbation scenarios where external events and processes such as natural geological and climatic phenomena, initial EBS defects and future human activities were examined.

FEPs that could generate isolation failure scenarios, such as direct human intrusion scenario and scenarios associated with natural phenomena were screened out (e.g. on the basis that they could be excluded or reduced to low likelihood by siting). Nevertheless, some "what if?" analyses have been carried out to illustrate the magnitude of potential consequences, and thus the importance of siting the repository in a suitable environment.

Comparing the scenarios considered in H12 with those considered in recent SA reports in other countries (e.g. Nagra, 1994b; Posiva, 1999), it was concluded that no significant scenarios have been overlooking that would be relevant to disposal in Japan.

Models and databases

Based on a list of feasible scenarios, models which simulate relevant phenomena in detail, together with associated databases, were established in order to quantify selected scenarios. Models were developed to simulate the evolution of the EBS and subsequent radionuclide migration in the rock surrounding the buffer material. These models are more detailed and realistic than those used in the H3 assessment and thus improve the understanding of key processes. The same can be said of the corresponding databases.

Models and datasets for near-field and geosphere modelling have been derived from, and tested against, the results of a laboratory and field experimental programme in Japan. In particular, use has been made of engineering-scale experiments at the ENTRY facility, experiments using radionuclides at the QUALITY facility and geoscientific investigations, mainly in Tono and Kamaishi (these studies should be seen as an illustration of the application of specific concepts, models and databases rather than an evaluation of potential sites). Extensive use has also been made of international scientific literature and international validation projects (e.g. NEA and SKI, 1994).

The main transport model used to represent EBS performance is based on one-dimensional, diffusive transport with linear, reversible and instantaneous sorption (processes which were taken into account in both H3 and H12). In H12, shared solubilities and precipitation of isotopes of a particular element are also accounted for during nuclide migration through the buffer. The lifetime of the overpack is assumed to be 1000 years. This represents the minimum duration determined from realistic modelling of the processes that may lead to overpack degradation and failure, as discussed in section 4.5.1. The long-term dissolution rate obtained experimentally for glass dissolution (1×10^{-3} g m^{-2} d^{-1}) (Ohe et al., 1991) is used in H12 at the waste glass-bentonite interface, supported by the accumulation of data and information from laboratory experiments and natural analogue studies. In H3, a more conservative approach was used based on a steady-state model for transport of silica through the buffer (see section 4.5.2). The radionuclides released at the outer boundary of the buffer are assumed in H12 to be instantaneously mixed within the EDZ. The groundwater flow rate in the EDZ is calculated using a three dimensional fracture network model, where the EDZ is modelled as a highly permeable continuous feature surrounding the emplacement tunnel. In H3, a simple zero-concentration boundary condition at the outer buffer surface was assumed.

H3, as well as several other contemporary safety assessments (e.g. SKB, 1992; Nagra, 1994b) considered a single fracture or channel to be representative of all transport paths within the host rock. To more realistically assess the performance of the host rock surrounding the repository, H12 considered transport along a set of representative channels, taking into account the heterogeneity of real fractures and channels with respect to transmissivity. In this one-dimensional multi-pathway model, the distribution of transmissivities is discretised, with each model pathway representing a set of channels of similar transmissivities (Fig. 2.4.2).

Advection and dispersion, matrix diffusion, sorption onto surfaces within the rock matrix and radioactive decay are taken into account in the modelling of transport within a single channel. It has been confirmed by numerical modelling that the multi-pathway model conservatively approximates nuclide transport in a more complex, stochastically generated three-dimensional fracture network (Ijiri et al., 1999), whereas the single fracture model may give unconservative results.

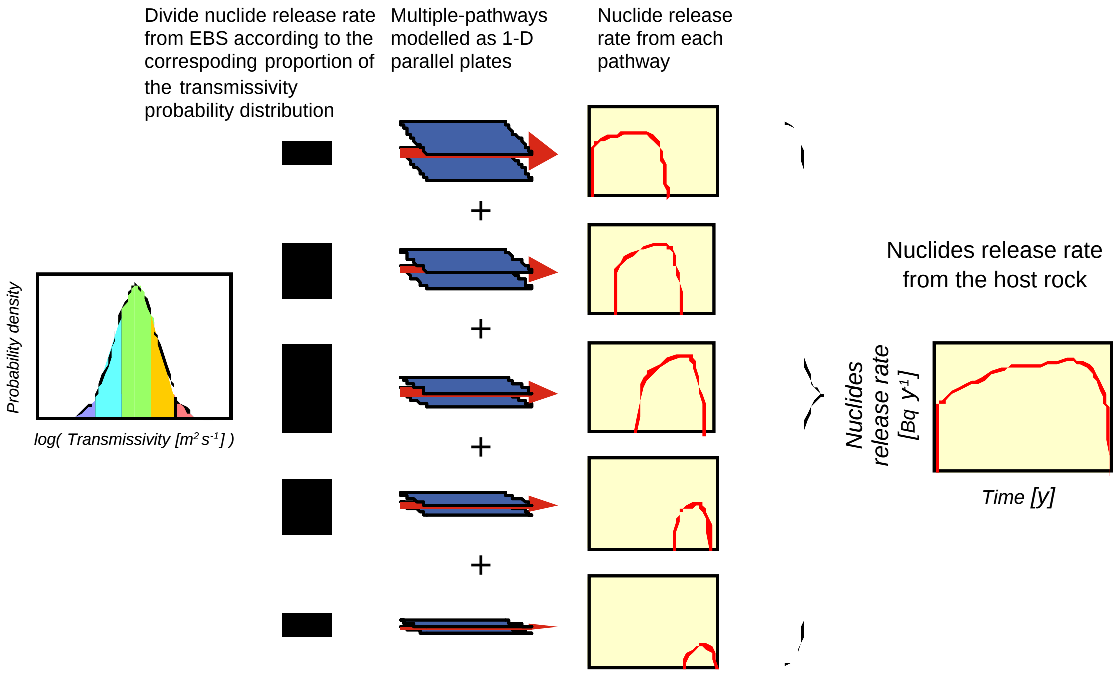


Fig. 2.4.2: Conceptual illustration of a one-dimensional multiple pathway model

In H3, it was assumed that at an arbitrary assessment point, groundwater containing radionuclides is diluted by uncontaminated groundwater before a dose calculation, using a simple dose conversion factor, is made to provide an estimate of the level of safety. In this case, the distance from the engineered barriers to the assessment point in the geosphere and the volume of water for dilution were considered as variable parameters. In H12, radionuclide releases from the waste packages are assumed to flow towards a single, major water-conducting fault located downstream from the repository. All radionuclides released from the repository are assumed to migrate upwards through this fault to a shallow aquifer which, in turn, discharges to a river. Significant dilution occurs as only a small amount of groundwater from the aquifer enters the river. Retardation is assumed to occur in the major water-conducting fault but not in the aquifer.

For biosphere modelling, H3 did not include a detailed evaluation of the consequences of radionuclide releases in terms of transport through the food chain and human uptake. In H12, more complex modelling was established to assess not just dilution in the biosphere but also the potential for transport and accumulation of radionuclides in the environment (Baba et al., 1999). No attempt was made to model the evolution of the surface environment and the lifestyles of future generations, due to uncertainties that are inherently irreducible. Rather, certain sets of assumptions were made about these aspects of biosphere modelling, giving rise to stylised representations of the biosphere for dose calculations ("Reference Biospheres", e.g. BIOMASS, 1999). The biosphere model represents the components of the surface environment using compartments between which fluxes of material (solid/water) and radionuclides are defined by transfer factors. A range of exposure pathways via which radionuclides could enter the food chain are also defined, along with uptake and concentration factors,. The resulting dose (from both ingestion and external irradiation) to a hypothetical critical group is then calculated. Parameters describing the processes in this system are based on estimates of present-day environmental and lifestyle conditions.

Computer codes used in all parts of the SA have been verified against similar codes used in other national programmes and, when available, analytical solutions in order to minimise the possibility of programming errors. All the modifications and corrections to the computer codes have been recorded systematically using software specially developed for this purpose.

2.5 Integration of R&D in safety assessment

The foregoing section, in describing the development from H3 to H12 of the disposal concept and the safety assessment models, databases etc., also gives an overview of the way in which R&D results have been integrated in the H12 assessment. In summary:

- Improvement of databases has allowed improved calculations of important processes, particularly in the near field. This, in turn, has been used to justify modifying design of the EBS, resulting in a reduction in thickness of both steel overpack and bentonite buffer. Improved knowledge of the properties of bentonite and bentonite/sand mixtures has also allowed the buffer specification to be changed from pure bentonite to a bentonite/sand mixture
- Much of the new information has resulted from in-house laboratory programmes set up during and after H3: from sorption and diffusion experiments for bentonite and potential host rocks in the ENTRY and QUALITY (active) facilities to in situ experiments at the Kamaishi and Tono test sites (e.g. Fujita et al., 1994; Ota et al., 1997 and 1999; Tanai et al., 1996a and 1996b; Oda et al., 1999), and international collaboration (e.g. at the Grimsel underground rock laboratory in Switzerland or Äspö in Sweden)
- In the area of scenario development, a key requirement has been sufficient knowledge to form the basis for the screening of FEPs in order to focus the assessment on relevant scenarios. R&D from a wide range of disciplines and sources is required for such a process, for example: geological understanding to assess FEPs

arising from the geological environment such as volcanism, faulting and uplift; engineering and rock mechanical understanding to assess both long- and short-term effects on the host rock arising from the construction and presence of the repository (e.g. Aoki et al., 1999)

- Information from natural analogue research programmes has provided support for the use of a long-term leach rate for the waste glass (e.g. Arai et al., 1989; Yusa et al., 1991; Kamai et al., 2000) and also for the iron corrosion rate for the steel overpack
- Improved understanding of groundwater flow in fracture systems, and particularly the impact of fast pathways, has not only resulted in a more realistic conceptual model for the geosphere in H12 but has also allowed assessment of the validity of the one dimensional SA model by comparison to more complex, stochastically generated three dimensional fracture networks (e.g. Yoshida et al., 1994a and 1994b)
- The development of a biosphere model for H12, which allows a wide range of potential radionuclide uptake processes and exposure pathways to be assessed, means that future assessments can be made site-specific by taking account of local factors such as local climate, land use and population life styles.

As processes can be modelled with in greater detail due to improvement in databases (and models), it is easier to identify those processes which are critical to the provision and maintenance of safety in the repository system. This in turn provides guidance for future R&D work priorities.

3 THE H12 SAFETY CASE IN PERSPECTIVE

P.A. Smith

3.1 The safety functions of the Japanese disposal system

Repositories for radioactive waste are sited and designed such that they will provide containment for prolonged periods, with any eventual releases to the biosphere being at levels that provide no threat to human health. Siting and design should ensure that there is a low probability that any event or process could significantly undermine the safety of the overall system. The probability of such a "single-mode failure" is minimised through the use of passive barriers with multiple safety functions.

The safety functions provided by the various components of the disposal system considered in H12 and their interaction are summarised in Table 3.1.1 (a more extensive summary is given in Tables 5.1-1 and 5.1-2 of H12 Supporting Report 3).

The period of containment within the overpack, combined with the slow release of radionuclides from the waste by dissolution of the glass matrix, transport through the bentonite and transport through the rock means that there is sufficient time for many radionuclides to decay to insignificant levels within these barriers, before reaching the human environment.

Tab. 3.1.1: The safety functions provided by the various components of the H12 disposal system

The engineered barrier system (EBS)
Complete containment by the overpacks for at least 1000 years
Very slow release from a stable glass waste matrix following breaching of overpack
Limitation of groundwater inflow, and filtration of colloids, microbes and organic matter by the bentonite buffer
Low solubility of key radionuclides in bentonite porewater (with redox buffering provided by overpack corrosion products)
Slow, diffusion-dominated radionuclide transport through the buffer, with retardation by sorption on mineral surfaces
The natural barrier
Provision of a suitable environment for the EBS (physical/chemical stability, low groundwater flow, suitable groundwater chemistry)
Slow radionuclide migration by slow advection through networks of water-conducting features in fractured media (and through rock pores in porous media, although the H12 SA assumes all rock types considered can be treated as fractured media)
For fractured media, retardation by matrix diffusion
Retardation by sorption on mineral surfaces

Fig. 3.1.1 illustrates this, showing a measure of the total radiotoxicity[17] of the nuclides contained within the different barriers as a function of time, as calculated for the H12 Reference Case. The figure indicates that:

17 The radiotoxicity index is here defined as the hypothetical dose from the ingestion of all nuclides in one year, divided by a reference dose rate of 0.1 mSv a^{-1}

- Radiotoxicity is entirely contained within the glass matrix until the overpacks are breached at 1000 years following emplacement, by which time the radiotoxicity has declined by one order of magnitude
- After 10^4 years, the radiotoxicity of each waste package is still contained predominantly within the glass matrix, and has declined to less than that of 10^6 m^3 of granite with 10 ppm uranium.
- Between about 10^5 years and 10^7 years, radiotoxicity is contained predominantly within the EBS, either sorbed onto bentonite or in the form of solid precipitates that are immobile in the small pores of the buffer
- Only at times of around 10^7 years is a significant proportion of the residual radiotoxicity transferred to the geosphere, by which time it has decayed by five orders of magnitude
- At no stage is a high proportion of the radiotoxicity contained within the biosphere.

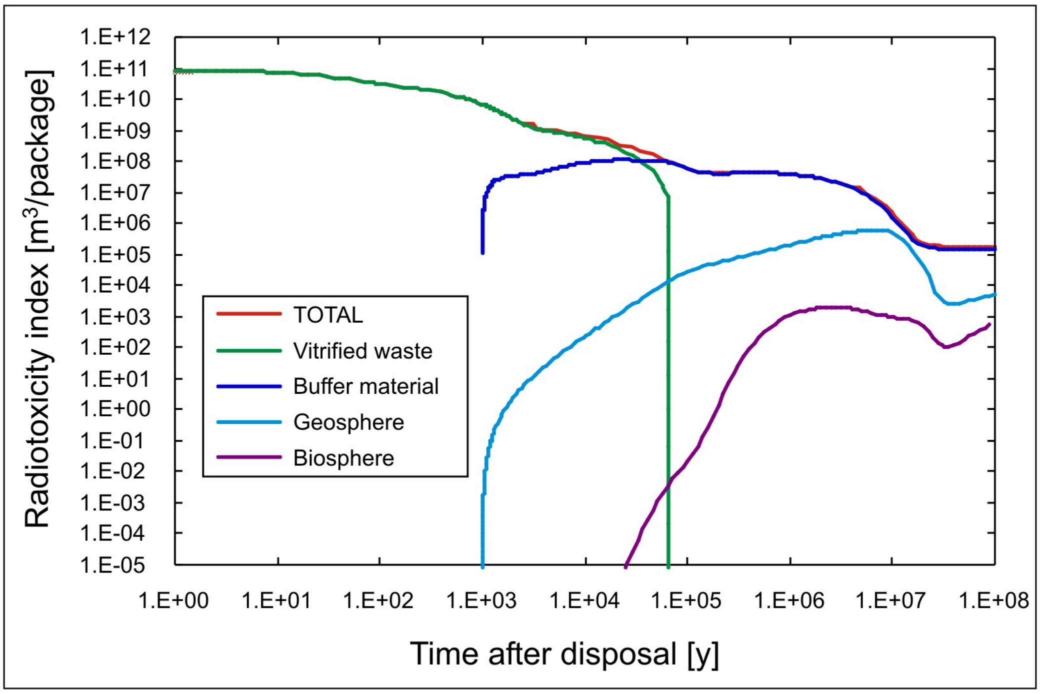


Fig. 3.1.1: Radiotoxicity index as a function of time for the H12 Reference Case, showing containment of radiotoxicity in the various components of the disposal system

3.2 The strategy for the making of the safety case

The safety case can be defined as the set of arguments that are used to show that a disposal system will meet the relevant radiation health protection standards (e.g. NEA, 1999b). In general, the safety case consists of:

- An assessment of the performance of the disposal system
- Evidence that supports the findings of this assessment
- An evaluation of the significance of any uncertainties or open questions.

Comparing different safety assessments carried out internationally, emphasis is placed on different safety functions, depending on the site and concept selected, the stage of repository planning and development and other programme- or project-related constraints (Table 3.2.1). In particular:

1) At a given stage in repository planning and development, a performance assessor will have the highest level of confidence in some safety functions, whereas others may be affected by significant uncertainties or open questions (section 3.3). The strategy for the making of a safety case may change as repository planning and development proceeds, and more information (e.g. on site-specific characteristics) becomes available.
2) For a particular site and design, certain safety functions are intrinsically more effective, or easier to demonstrate, than others.

To illustrate this second point, Figure 3.2.1 shows the relative quantities of materials (by volume, per waste package) in the EBS designs for H12 and TVO 92[18]. The Finnish case is chosen here to provide contrast with H12, being based on a relatively thin, corrosion-resistant, composite copper-steel overpack, and deposition-hole emplacement, with a relatively narrow bentonite buffer separating the overpack from the surrounding host rock. The Finnish safety case, as well as those of the Swedish implementer (SKB) and regulator (SKI) focus primarily on complete containment for very long times by the overpack. Indeed, in order to carry out quantitative SAs, initial overpack defects have been postulated, even though it is currently not possible to quantify the likelihood of such defects. In contrast to H12, where corrosion products from the massive steel overpack ensure reducing conditions in the buffer, an important concern in the Finnish and Swedish analyses is the possibility of events and processes that may give rise to oxidising conditions. In this respect, oxidants arising from radiolysis of water, which are greater in the case of spent fuel than vitrified HLW, are of more concern in these analyses than in H12 even though some redox buffering due to corrosion of the iron insert after breaching of the copper overpack can be expected.

18 TVO 92 is used, rather than the more recent Finnish SAs, since the overpack design is generic in TVO 92. In TILA 99, for example, the overpack design is adapted to the different fuel assemblies arising from the Loviisa and Olkiluoto reactors resulting a significant difference in size and mass of the waste packages

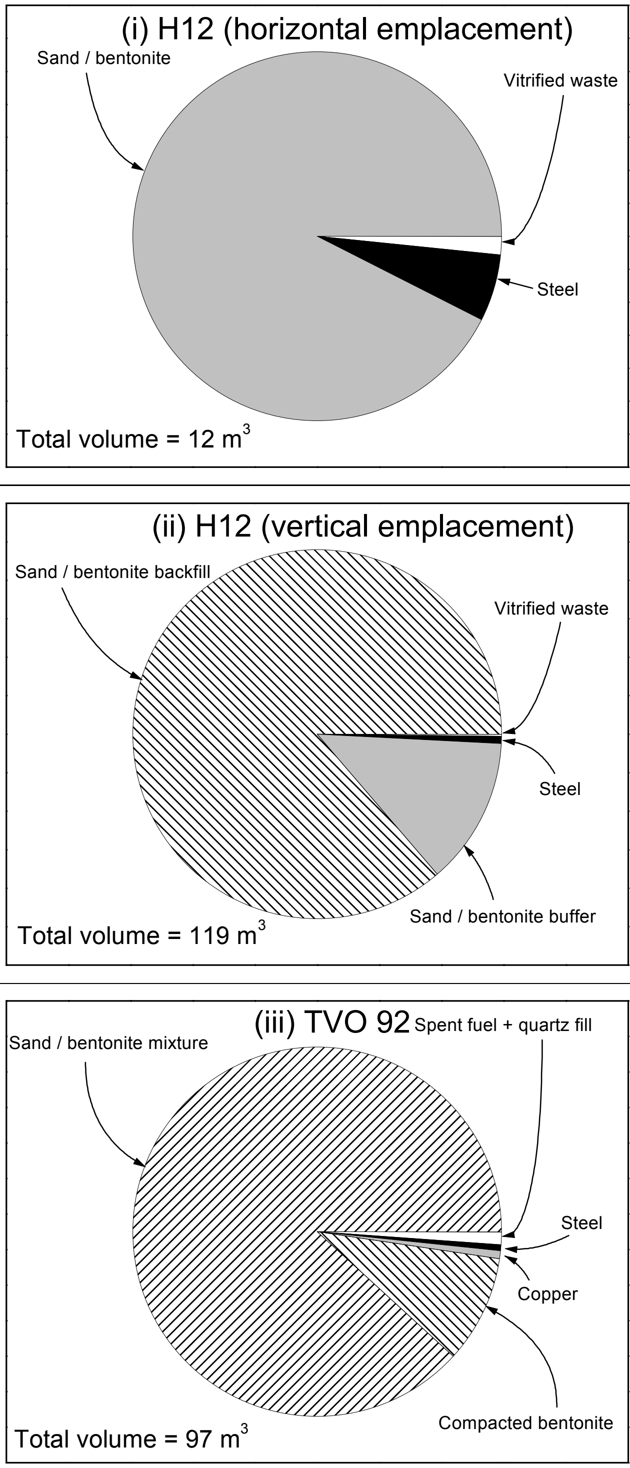


Fig. 3.2.1: Relative quantities of materials (by volume, per waste package) in the engineered barrier designs for (i), H12 horizontal emplacement, (ii), H12 vertical emplacement and (iii), TVO 92 (see Appendix A for explanation)

Tab. 3.2.1: Contrasting emphasis on different safety functions in various safety cases.

Emphasis on near field	
Safety function	**Safety assessment**
Longevity of intact overpack (complete containment, e.g. by copper overpack)	SKB 91 and subsequent SKB assessments SITE 94 TVO 92 and subsequent Posiva assessments AECL EIS (assessment of alternative system)
Low solubilities and/or high sorption provided by bentonite backfill	H12 Kristallin-I
Emphasis on geosphere	
Safety function	**Safety assessment**
Complete containment by host rock (salt)	USA WIPP PSE Gorleben
Long transport times through low-permeability, non-fractured crystalline rock	AECL EIS (assessment of reference system)
Long transport times through low-permeability, non-fractured clays	UPDATING 1990
Unsaturated, low-flow environment	Yucca Mountain Project (YMP)

In the case of H12, the results of the SA show that both the EBS and the natural barrier contribute to the overall performance of the disposal system. Nevertheless, even when no credit is taken for the geological transport barrier, the system is sufficiently robust that the calculated dose maxima remain low, as long as the host rock continues to provide a suitable environment for the EBS, and provided the repository facilities are adequately designed and constructed (e.g. such that backfilling is complete and repository seals are effective). The EBS, therefore, plays a key role in the safety case.

This is important given the current uncertainties in the geological database for Japan, as well as the uncertainties inherent in the use of generic geological data in H12. Emphasis on the EBS is supported, for example, by experimental studies and scoping calculations which indicate that the key safety functions of the bentonite buffer will be retained for a long period, provided an appropriate geological environment is selected (Kanno and Wakamatsu, 1991; Kanno et al., 1999).

Although the H12 safety case emphasises the EBS, all safety functions are considered in the safety assessment, in order that the safety case can be said to provide "defence in depth" – i.e. if one safety function were to operate less effectively than expected, safety would, in any case, be guaranteed by other safety functions[19].

This is illustrated in Figure 3.2.2, which shows:

A. The percentage decay of the inventories of radionuclides in the vitrified HLW before overpack breaching at 1000 years

19 "Defence in depth" arising from the multi-barrier disposal concept should not be confused with the term applied to reactor safety where independent and redundant barriers can be engineered

B. The percentage by which the releases from the HLW decay during transport through the near field

C. The percentage by which near field releases decay during migration along a transport path through the geosphere which, in H12, consists of 100 m through undisturbed rock and 800 m through a major fault.

The degree of decay during near field and geosphere transport is estimated using simple steady state "insight models", which are described in Appendix B.

The radionuclides shown in the figures are the "safety-relevant" radionuclides identified in H12 (see, for example, Table 5.3.1-2 in H12 Supporting Report 3), excluding some shorter-lived daughter radionuclides in decay chains, the releases of which are determined mainly by ingrowth from their parents rather than by their own transport properties. Many radionuclides that decay to insignificance during containment in waste packages are excluded from the list of safety-relevant radionuclides, and so are not shown in the figures.

Figure 3.2.2 illustrates that slow near-field and geosphere transport processes are highly effective in ensuring attenuation by decay of the releases of many safety-relevant radionuclides. Some radionuclides (e.g. Tc-99 in the case of H12) are sufficiently long-lived and / or weakly sorbing that they escape from the near field without substantial attenuation by decay. Some of these are attenuated by decay during geosphere transport. Others, such as Cs-135 and Pd-107, have the potential to reach the biosphere without substantial decay. The release rates of these nuclides are limited by solubilities in the EBS (although not in the case of Cs), the dissolution rate of the waste form, diffusion in the buffer and the water flow rate through the rock.

Figure 3.2.3 gives similar plots for the Finnish TILA 99 SA for spent fuel. I-129 is included in Figure 3.2.3 because of its safety relevance for this waste form. In the TILA 99 concept, if, as expected, the overpack remains unbreached for 10^6 years or more, all but a few nuclides decay to insignificant levels by the time breaching occurs. If, however, the overpacks were to be breached earlier than this, most radionuclides would still decay to insignificance, especially during geosphere transport. This is an example of the "multi-barrier concept", whereby if one barrier were to operate less effectively than expected, most radionuclides would then decay in another.

A comparison of Figures 3.2.2 and 3.2.3 shows the greater effectiveness of the overpack in the case of TILA 99 and the greater effectiveness of the near field as a transport barrier in the case of H12 (note the change of scale in graph (B) in Fig. 3.2.3). This is due principally to the greater thickness of bentonite around the waste packages in the H12 design, as well as differences in the degree of sorption, as discussed further in Chapter 4.

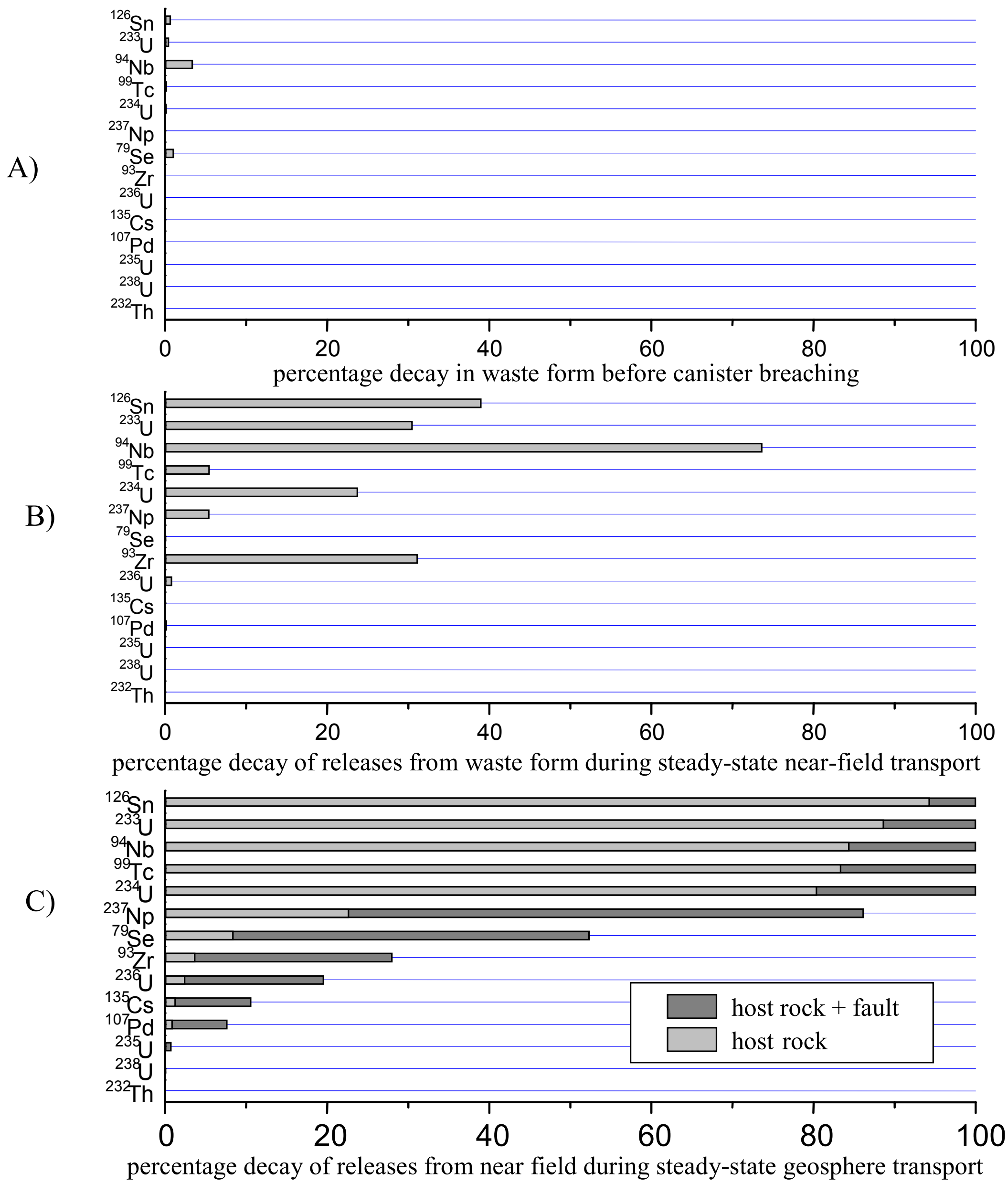


Fig. 3.2.2: The percentage decay of the inventories of radionuclides in the vitrified waste form before overpack breaching in the H12 concept, and the percentage by which the releases from the waste form decay during transport through the near field, and decay further during migration through the geosphere.

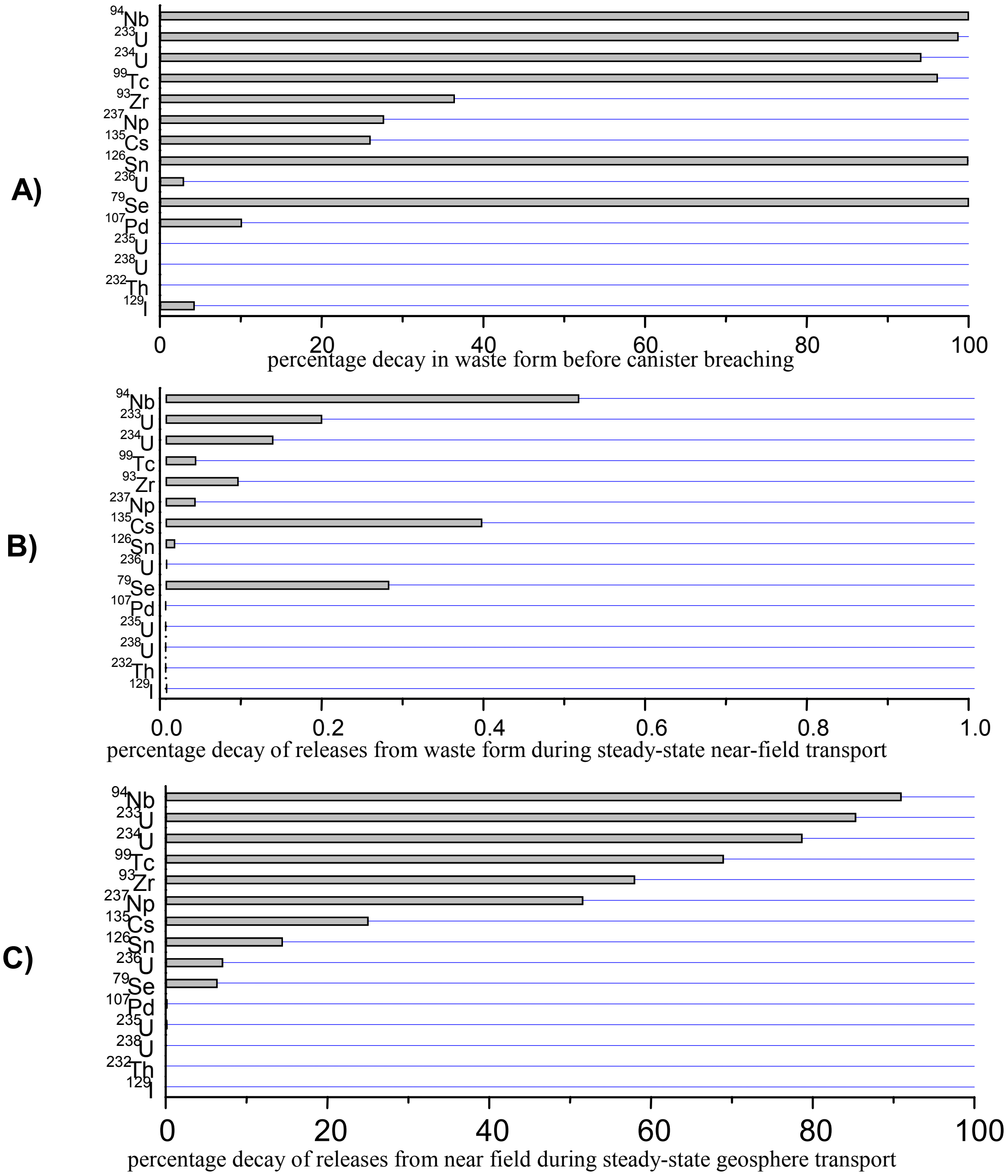


Fig. 3.2.3: The percentage decay of the inventories of radionuclides in the vitrified waste form before overpack breaching in the TILA 99 concept, and the percentage by which the releases from the waste form decay during transport through the near field, and decay further during migration through the geosphere.

3.3 Significance of uncertainties and open questions

According to an NEA position paper on confidence building (NEA 1999b), a safety case should acknowledge the existence of any unresolved issues and provide guidance for work to resolve these issues in future stages of repository planning and development.

In H12 (Table 7.2-1 in Project Overview Report), in order to reach the stage of characterising potential candidate sites, R&D requirements have been identified in the areas of:

- Technologies for determining the stability of the geological environment
- Techniques for characterising the geological environment
- Engineering technologies
- Techniques for detailed repository design
- SA methods, models and data.

This is put in context in Table 3.3.1 which identifies specific open issues and R&D requirements related to methods, models and data that have been identified in H12 and other selected SAs. The comments made in the table regarding the relevance to the safety case illustrate how open issues identified as being important (and thus R&D requirements) depends both on the disposal concept and the stage of repository planning and development.

Tab. 3.3.1: Open issues and R&D requirements related to methods, models and data that have been identified in various, selected safety assessments

Issues and requirements	Comments/relevance to safety case	SA
Databases (general):		
Completion of radionuclide inventories with data relevant to long-term SA and requirement for additional data regarding physical/ chemical waste characteristics (presence of organics, etc.)	Reflects preliminary nature of Boom Clay repository studies in 1990	UPDATING 1990
Expansion of thermodynamic databases for SA transport modelling	Reflects, for example, importance of solubilities in limiting releases in H12	H12, UPDATING 1990
Reduction of uncertainty in sorption data for EBS	Reflects sensitivity in the case of weakly sorbing radionuclides	SITE 94

Tab. 3.3.1 continued

Issues and requirements	Comments/relevance to safety case	SA
Near-field models:		
Improved models for evaluating overpack lifetime	In the case of SITE 94, this reflects the potential of the composite copper/steel overpack for complete containment for hundreds of thousands of years, and the need to understand events/ processes that could compromise this (e.g. common-cause failures under glacial conditions)	H12, SITE 94
Improved understanding of glass fracturing	Glass dissolution depends on surface area which is increased by fracturing	UPDATING 1990
Improved understanding of the impact of oxygenated waters on the buffering behaviour of bentonite (especially mobility of ^{99}Tc)	Reflects concern over penetration of oxygenated waters in glacial scenarios – especially relevant for a copper-clad overpack, where iron corrosion products cannot be relied upon to ensure reducing conditions	SITE 94
Development of a more realistic model for extrusion of buffer materials	Reflects the requirement in the H12 safety case that the buffer maintains its physical characteristics (colloid filtration, etc.)	H12
Development of models/data relevant to the hydrogeological effect of the excavation-disturbed zone	The EDZ has the potential to provide a "fast path" linking the near field and biosphere if repository seals cease to operate effectively	Kristallin-I
Improved understanding of effects of very saline groundwater, and hyperalkaline fluids from concrete, on buffer and backfill.	Hyperalkaline fluids from co-disposed long-lived ILW (TRU) wastes or cement grouts and seals.	Kristallin-I (in the case of hyperalkaline fluids), TILA 99
Development of understanding of gas generation and release, and bentonite gas permeability	Reflects the potential of these processes to perturb the properties of the buffer	Kristallin-I, TILA 99
Development of understanding of the mechanical behaviour of the overpack-buffer-backfill system.	Reflects concern regarding potential movements of the overpack (sinking/lifting), which could lead to reduction of buffer thickness between overpack and rock and possible damage to the overpack.	Kristallin-I. TILA 99

Tab. 3.3.1 continued

Issues and requirements	Comments/relevance to safety case	SA
Geosphere models:		
Development of advanced hydrological and radionuclide transport models and associated databases, and application to specific geological environments	Reflects the desire to reduce conservatism when hydrogeological / transport models are applied to actual sites	H12
Development of improved understanding of small-scale structure of water-conducting features – e.g. effects of fracture coatings, distribution of matrix porosity	Reflects sensitivity of consequence analysis to assumed small-scale structure	SITE 94, Kristallin-I
Development of understanding of the possibility of irreversible sorption on groundwater colloids	Has the potential to reduce the retarding effects of matrix diffusion and sorption on geosphere transport (although, if the colloids are filtered during transport, irreversible sorption of radionuclides may reduce releases)	Kristallin-I
Density driven flows and, in the case of TILA 99, more detailed studies of flow of saline groundwater at coastal sites	In TILA 99, the modelled system is sensitive to the combination of very high flow and saline water chemistry	Kristallin-I, TILA 99
Biosphere models:		
Application of biosphere models to site-specific conditions	Reflects sensitivity of consequence analysis to biosphere assumptions (principally dilution)	H12
Radon pathways and doses	The treatment of Rn-222 at the geosphere/biosphere interface, and possible dose pathways due to radon and its short-lived daughters, require particular attention	Kristallin-I
Other issues		
Quantification/ estimation of likelihood/ consequences of inadvertent human intrusion scenarios	Have the potential to generate alternative pathways for release to the human environment	H12, Kristallin-I

In its review of developments in the last decade, the NEA (NEA 1999a) identified the following additional areas where more work on safety assessment is seen as needed, or at least desirable:

- Sorption on overpack corrosion products, which may contribute significantly to the safety of some repository concepts, but is not, in general, considered to be supported by sufficient data to include quantitatively in safety assessments
- The treatment of climatic and geological events and changes, although advances have been made in the quantitative assessment of the impact of climatic change (e.g. in the SITE 94 safety assessment) and, in the USA, initial attempts have been made to quantify the effects of climate change, volcanic and seismic events on system performance
- The treatment of coupled phenomena [20] (thermal, chemical, mechanical and hydrological), that may affect, for example, the early phase of heating and resaturation of a buffer, and also influence its long-term performance.

It is interesting to note that, although confidence in the completeness of the phenomena analysed in a SA is a key element of any safety case, none of the SAs identify this as a priority area; R&D requirements relate principally to consequence analysis rather than scenario development. The only significant exception to this is work being carried out on the hyperalkaline plume development, largely for cementitious repositories for disposal of low and intermediate level wastes but also of relevance where long-lived ILW is co-disposed with HLW (e.g. Projekt Gewähr, Nagra 1985). Such work could lead to scenario development.

20 Large-scale experiments are underway to develop understanding and test models of these phenomena (e.g. large-scale heater tests at Yucca Mountain and the FEBEX experiment at the Grimsel Test Site; large-scale laboratory experiments in the JNC ENTRY programme, Japan and the FEBEX mock-up experiment of ENRESA in Spain).

4 THE APPROACH TO SAFETY ASSESSMENT IN PERSPECTIVE

P.A. Smith, R.A. Klos, F.B. Neall, P.Wersin, E. Curti

4.1 Introduction - the approach to safety assessment

In spite of differences in the detail of SA methodology between national programme, certain broad steps are common to most safety assessments, including H12. These steps, described in NEA (1999a), consist of:

(a) Scenario development:

This is the definition of scenarios, each representing (in a simplified manner) the evolution of the disposal system in response to particular system features, events or processes (FEPs). The scenarios provide the basis for specific cases to be considered, either quantitatively or qualitatively, in the safety assessment.

In Section 4.2, the strategy adopted in H12 for the selection of FEPs for quantitative analysis is compared to that of other assessments. In Section 4.3, the scenarios that are considered in H12 are compared to those addressed in other assessments.

(b) Consequence analysis:

This is the application of methodologies, models, databases and codes for the quantitative evaluation of repository performance for selected scenarios.

A key aspect of consequence analysis is the evaluation of the influence of uncertainty. Various types of uncertainty are identified (see section 4.4 for further description of the categorisation of uncertainty, and the deterministic approach to sensitivity analysis adopted in H12) - in the scenarios developed for evaluation, in the conceptual models to describe processes, and in the data used to quantitatively evaluate the performance of the repository system. The aim of this analysis of the sensitivity of the results to various types of uncertainty is to investigate whether repository performance is unacceptably compromised by any specific uncertainty. Clearly, if performance is shown to be so compromised, further work will required to reduce the specific uncertainty involved or to re-design the system so as to make it less susceptible.

In H12, a Base Scenario and a number of Perturbation Scenarios are defined. The treatment of near-field, geosphere and biosphere features and processes in the Base Scenario is compared to that in other assessments in Sections 4.5, 4.6 and 4.7, respectively.

(c) An assessment of the available safety margins:

The long-term consequences of the evolution of the repository system, evaluated in safety assessment, may be expressed in terms of various indicators, such as dose to a critical group or radiotoxicity (IAEA 1994). The findings of the H12 safety assessment and the performance indicators used are discussed in Chapter 5.

4.2 The strategy for the selection of FEPs for quantitative analysis

In H12, the selection of FEPs for quantitative analysis proceeded by:

- Preparing a comprehensive FEP list
- Repeated screening of FEPs according to a set of defined criteria.

This general procedure is common to most safety assessments, although the screening criteria are tailored to the disposal system under consideration and to the aims of the assessment.

The sources that were drawn upon to generate the H12 FEP list, and the FEP lists of other selected safety assessments, are presented in Table 4.2.1. In addition to these sources, organisations generally draw upon expert opinion (from both inside and outside the organisation) to ensure that the FEP list represents the current state of scientific understanding of the disposal system and also of the events and processes that may impact on its evolution.

Tab. 4.2.1: Examples of sources that were drawn upon to generate the comprehensive FEP lists in different safety assessments

Source of FEPs	Safety assessment
Generic FEP lists, prepared within international collaborative projects (IAEA, 1985; NEA, 1992; NEA, 1999c)	H12, Kristallin-I, TILA 99, MOL 94 (update of scenarios after UPDATING 1990), SITE 94
Earlier exercises by the organisation carrying out the assessment	H12 (draws upon H3), Kristallin-I (draws upon Project Gewähr), TILA 99 (draws upon TVO 92 and TILA 96)
Concept-specific exercises carried out either independently by, or in collaboration with, other organisations	SKB 91 (draws upon joint study by SKI and SKB) TILA 99 (audit against FEP databases developed by Swedish organisations - SKI and SKB)

Examples of screening criteria are presented in Table 4.2.2. The criteria are organised into the categories:

Site:

Criteria that exclude FEPs that are either impossible or irrelevant to the host-rock types of interest, selected siting areas or selected site.

Repository design:

Criteria that exclude FEPs that are either impossible or irrelevant to the selected design.

Scope of the assessment:

Criteria that exclude FEPs the consideration of which is beyond the defined scope of the assessment.

Preliminary assessment:

Criteria that exclude FEPs which are judged (by qualitative arguments or scoping calculations) to have insignificant impact on long-term safety, either in terms of likelihood of occurrence or in terms of consequences.

Tab. 4.2.2: Screening criteria used to narrow the range of FEPs that need to be considered in performance-assessment cases

Screening criteria	Examples of screened-out FEPs
Site:	
Irrelevant to host-rock type(s)	Salt diapirism (Kristallin-I)
Irrelevant to local and regional surface environment	Estuarine and marine FEPs (Kristallin-I)
Irrelevant to geographical location	Continental ice sheet (H12) Sea-level change (Kristallin-I)
Repository design:	
Irrelevant to waste-form and packaging	FEPs relevant to direct disposal of spent fuel (H12, Kristallin-I)
Irrelevant to repository design	FEPs relevant to cementitious backfill (H12, Kristallin-I)
Scope of the assessment:	
Irrelevant providing a repository is appropriately sited	Bentonite erosion (Kristallin-I) Improperly emplaced buffer (SITE 94)
Irrelevant providing a repository is appropriately designed and constructed (deviations from the Design Basis)	Repository left unsealed (Kristallin-I) Initial defects in engineering (H12) [1]
No consideration of global and regional disasters	Nuclear war, meteorite impact[2] (H12, Kristallin-I)
No consideration of malicious acts	Terrorism (H12, Kristallin-I, MOL 94)
No consideration of deliberate intrusion	Recovery of wastes (H12, Kristallin-I)
No consideration of evolution of human society and technology	Development of new technologies, medicine, etc. (H12, Kristallin-I)
No consideration of evolution of human beings and other species	Changing radio-sensitivity of human beings; new crop/animal species (H12, Kristallin-I)
Other FEPs ruled out by the scope of specific assessments	In SITE 94, human intrusion was considered to be fundamentally different in nature from other phenomena, and a requirement was identified for a separate position on the philosophy for considering human intrusion in a regulatory context. In Kristallin-I, human intrusion was excluded specifically under Swiss regulations. Retardation in tunnels and shafts was not taken into account in H12 as no safety function was as assigned to tunnel backfill, seals and grouting in the SA.
Preliminary assessment:	
Low likelihood of occurrence [3]	Criticality[4] (H12) Waste package sinking (H12) Meteorite impact[1] (MOL 94)
Consequences can be shown to be insignificant, e.g. by simple, scoping calculations	Release of radioactive gases (Kristallin-I) Chemical alteration of host rock (H12)

Notes:

1 It is expected that quality management procedures will detact initial defects in the engineered components of the repository. however, perturbation scenarios assuming incomplete welding of the overpack, poor backfilling of the tunnels and defective plugs were analysed following examples considered elsewhere (e.g. AECL, 1994; Nagra, 1994a; Vieno and Nordman, 1999)

2 Meteorite impact can be excluded either on the grounds that an impact substantial enough to perturb the repository would have far more significant non-radiological consequences, or on the grounds of low likelihood.

3 In MOL 94, for example, events are discarded if their probability of occurrence is estimated to be less that 10^{-8} per year. In H12, FEPs were screened using more qualitative "expert judgement"

4 Criticality can also be excluded on the grounds of insignificant consequences in the case of H12 (HLW).

4.3 The scenarios considered in safety assessments

In H12, the FEPs selected for inclusion in consequence analysis are used to define:

- a Base Scenario[21]

and a set of:

- Perturbation Scenarios.

The Base Scenario considers many phenomena that are certain, or highly likely, to occur. The precise definition varies between assessments. The following overview is taken from p.8 in Marivoet (1994):

- The IAEA report on safety assessment (IAEA, 1985) says: "the observation that some phenomena are certain to occur leads to the concept of a "normal"" scenario, which consists of the most probable sequence of events following repository closure."
- …
- In the USA the EPA regulations (US EPA, 1985) speak about undisturbed performance: " Undisturbed performance" means predicted behaviour of a disposal system, including consideration of the uncertainties in predicted behaviour, if the disposal system is not disrupted by human intrusion or the occurrence of unlikely natural events."
- In the Canadian scenario analysis (Stephens and Goodwin, 1990) the concept "central" scenario is used: the central scenario describes the most probable complete mechanism by which waste materials may be released from the vault, traverse the geosphere, contaminate the biosphere and lead to radiation doses to humans. It contains as many factors as possible."

The term "Base Scenario", or even "Normal Evolution Scenario", is not, however, intended to imply that this scenario is viewed as a realistic representation of the actual evolution of the disposal system. In particular, the assumption that the surface environment remains unchanged indefinitely is certainly unrealistic. Rather, it provides

21 Other assessments define, for example, a Reference, Central or Base-case scenario. The precise definition of this scenario and, in particular, the degree to which it represents the "expected" evolution of the system varies between assessments.

a basis for illustrating system performance under certain well-defined assumptions about the processes involved. The perturbation scenarios illustrate system performance under alternative assumptions which give rise to additional possibilities for system evolution. The aim in defining these scenarios is to encompass all significant possible evolutions of the system. Quoting from SITE 94:

> Scenarios are not predictive devices, but are means of stimulating and disciplining the imagination so as to provide an organised way of *illustrating* possible future behaviour of the system and defining how such behaviour might arise.

In H12, the Base Scenario and most of the Perturbation Scenarios are classified as "Groundwater Scenarios": i.e. scenarios in which transport of radionuclides from the repository to the surface environment is mediated by transport processes in groundwater. "Isolation Failure Scenarios" are also defined in which the physical isolation of the waste is compromised either directly, for example by drilling into the repository, or indirectly, by uplift and erosion exposing the repository at the surface.

These "Isolation Failure Scenarios" are associated with FEPs which were screened out due to low probability or the possibility to avoid them by suitable siting and design. They were treated with informal "what if" calculations.

Some assumptions of the Base Scenario and of the Reference Scenario/ Base-case Scenarios in other assessments are given in Table 4.3.1.

There are some significant differences between assessments, for example:

(i) *The treatment of the overpack in assessments that consider copper overpacks (or composite copper-steel overpacks) in crystalline rock (e.g. TILA 99, SKB 91).*

All such assessments have come to the conclusion, or have implicitly assumed that, initially intact copper overpacks preserve their integrity for a very long time in the normal evolution of the repository. An initial defect is assumed in the Reference Scenario in order to allow an assessment of the performance of the other barriers[22].

However, it is not currently possible to quantify the probability of such defects. In the case of the steel overpacks of the H12 reference design, an initial defect is also considered, but only as a Perturbation Scenario.

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22 These assessments deal with the disposal of spent fuel. The possibility of initially defective overpacks is of particular relevance to spent-fuel disposal, because of the potential for rapid release of nuclides that have accumulated, for example, in the gap between the cladding and the fuel. This "instant release fraction" is not a concern in the disposal of vitrified HLW

Tab.4.3.1: Definition of the Base Scenario in H12, and comparison with Reference Scenarios considered in other assessments

Assumptions	Safety assessment
Engineered-barrier system (EBS):	
The evolution of the EBS proceeds such that it performs its design functions [1]	H12, Kristallin-I
It is assumed that there is either a single, initially-defective overpack, or, in the case of TILA 99, that the overpacks disappear after a certain time (10^4 years).	SKB 91, TILA 99, SITE 94
"the overall conclusion of the Base scenario is that the Cu overpack's isolating capacity is not threatened by mechanical or chemical stresses…. safety margins are great even in a 1 Ma perspective"	SR 97
Geological conditions:	
Current conditions persist indefinitely	H12, Kristallin-I, SKB 91, TILA 99
Response of groundwater flow and composition in response to climate change considered	SITE 94
Near-surface environment:	
Current conditions (climate, topography, etc.) persist indefinitely	H12, Kristallin-I, SKB 91, TILA 99
A deterministic description of the likely climate state at the hypothetical site (Äspö) is assumed, based on a climate-evolution model for Sweden	SITE 94

Note:
1: The lifetime of the overpack is assumed to be at least 1000 years, which precludes perturbations from radiogenic heat and radiolysis in the analysis of radionuclide dissolution and migration through the EBS. Following breaching, the overpack is conservatively assumed to offer no transport resistance; credit is, however, taken for the redox buffering effects of the overpack corrosion products.

(ii) The treatment of climate change in the SKI SITE 94 Central Scenario

The SITE 94 Central Scenario differs from the others listed on Table 4.3.1 in that it includes climate change, climate-driven processes and their impacts on the disposal system. The assessment, however, also considers a Reference Case (not considered to be a "scenario" in SITE 94) which describes " … the "internal evolution" of the repository system when it is not under the impact of changing external influences". It is thus, in practice, similar to the Normal Evolution and Reference Scenarios considered in other assessments.

The perturbation scenarios considered in different assessments are summarised in Table 4.3.2. It is interesting to contrast the situation in the Japanese programme, where it is expected that volcanism can be screened out of necessary perturbation scenarios by choice of a suitable site[23], with Yucca Mountain. Since the site at Yucca Mountain was chosen, the age of nearby volcanic cones has been found to be much younger than originally thought (CRWMS M&O, 1998). As a consequence, the safety assessment must consider the possible effects of volcanism, both eruptive centres within the repository and intersection of the repository by dykes. The treatment is fully probabilistic; the mean annual frequency of intersection of the repository (primary block) by a dyke is estimated to be 1 x 10^{-8} and that of occurrence of one or more eruptive centres within the repository to be 7 x 10^{-9} (CRWMS M&O, 2000). The analysis of dyke intrusion must consider the distance magma travels along (open) drifts, the number of waste packages affected and the thermal and mechanical environment experienced by the waste packages in order to calculate the failure rate. A similarly complex analysis is made for an eruption through the repository giving rise to ash clouds containing contaminated particles.

Tab. 4.3.2: The perturbation scenarios considered in different assessments

Perturbation scenario	**Safety assessment/ comments**
Defects in, or unexpected performance of, the engineered-barrier system:	
Incomplete sealing of the overpack	H12 (initial defects assumed in Reference Scenarios in TILA 99 and SKB 91)
Transport along tunnels and shafts[1]	Kristallin-I, MOL 94, ENRESA-1997, SITE 94, H12 - may be relevant if the long-term effectiveness of tunnel/shaft seals cannot be guaranteed
Oxidising conditions in the near field	TILA 99 – due to the effects or radiolytic oxidants, which is particularly relevant to the direct disposal of spent fuel in Cu overpacks
Expulsion of radionuclides from overpack by gas	TILA 99
Overpack sinking	Kristallin-I
Geological events and processes:	
Uplift and erosion	H12, Kristallin-I – particularly relevant in tectonically active regions such as Japan and Switzerland
Tectonically-induced seismicity[1]	SITE 94
Alternative exfiltration area (small valley)	Kristallin-I
Fault activation[1]	UPDATING-1990, MOL 94
Volcanism[1]	YMP – treats both volcanic eruption and igneous intrusion

23 In H12, isolation failure due to volcanism is treated by a "what if" calculation based on a low rate of new volcano formation (<10^{-7} a^{-1}) at the repository site. This led to the conclusion that radionuclide release from the repository by volcanic acitivity, expressed in terms of equivalent U-238 release, would not be significantly greater than the U-238 contained naturally in the lavas

Table 4.3.2 continued

Perturbation scenario	Safety assessment/ comments
Climatic events and processes:	
Sea level change	H12 – relevant to coastal sites (alternative geological environments were considered quantitatively, e.g. different hydraulic gradient, topography, saline/fresh water, and the effect of sea level change on erosion rates qualitatively, but no explicit consideration of sea level change).
Changes in temperature/ rate of precipitation	Kristallin-I - alternative, dry and humid climate states considered, UPDATING 1990 - reduction in precipitation rate considered, SITE 94 - warm, wet climate considered as alternative to Central Scenario MOL 94 – greenhouse effect considered
Development of permafrost	Kristallin-I
Sub-glaciation erosion	MOL 94 – consideration of substantial sub-glacial erosion, leading to disruption of the repository
Post-glacial faulting	TILA 99 - effects of a major post-glacial rock displacement breaking overpacks, bringing oxic water into the repository and creating a fast pathway to the biosphere
Glacial meltwater	TILA 99 – oxygenated glacial meltwater intruding into the bedrock beneath a warm-based ice sheet in the melting phase of the glacier
Future human activities:	
Deep groundwater well	H12, Kristallin-I, MOL 94, ENRESA-1997, SITE 94 - intake of contaminated well water considered
Borehole intersecting repository	H12 – effects considered of (i), influx of oxidising water from the surface and (ii), fast radionuclide transport path provided by borehole disturbed zone (also in WIPP[2])
Liquid waste injection into a fracture zone near the repository	SITE 94
Liquid waste injection into a poorly sealed shaft, combined with local well/mine pumping	SITE 94
Human impacts on the surface and on groundwater recharge	SITE 94
Mining impacts on the surface and on groundwater recharge	SITE 94
Exploratory drilling/ archaeological investigation	MOL 94 - examinations of cores extracted from within or around the repository
Exploitation of geothermal energy	Kristallin-I

Note:
1: Discussed in H12, but subsequently screened out, and thus not classified as a perturbation scenario. Volcanism, fault activation and borehole intersecting the repository are treated as isolation failure scenarios (informal "what if" calculations).
2 Waste Isolation Pilot Plant, New Mexico, U.S.A.

4.4 Treatment of uncertainty and development of cases for consequence analysis

NEA (1997) provides a review of methods for the treatment of uncertainty in safety assessments. This review brings together elements from ten safety assessments performed during the last decade and distinctions are drawn between the various types of uncertainty. These are well represented in the H12 report, indicating that H12 is directly comparable to contemporary applications.

The three classifications identified by NEA (1997):

- *Scenario uncertainty* which arises from limited knowledge of:
  - The evolution of slow processes such as chemical interactions between the EBS materials and groundwater
  - The timing and frequency of events that may affect the stability of the geological environment
  - Future human activities.
- *Conceptual model uncertainty* arises where understanding of a process to be modelled is not sufficient to discriminate between alternative descriptions (models) of the process. Model uncertainties can also arise due to errors in formulating models.
- *Parameter uncertainty* (spatial and temporal variability as well as genuine lack of knowledge) arises from:
  - Measurement errors
  - Interpolation of spatially heterogeneous geological properties
  - Extrapolation of experimental or natural analogue results to timescales or conditions relevant to SA.

These types of uncertainty are approached systematically in H12:

- *Scenario uncertainty* – by a clearly defined approach to the identification, characterisation and implementation of modelling scenarios. Each identified scenario involves a unique combination of FEPs which allows the influence of alternative FEPs to be assessed relative to the combination used in the Base Scenario;
- *Conceptual model uncertainty* – by the identification and analysis of equivalent conceptual representations of models within a given scenario. For example, where two equivalent candidate research models [24] exist for a process, a number of alternative calculational cases is defined so that the significance of differences between the models can be evaluated. At the level of a SA model, the approach used in H12 is to verify that simplifications of the corresponding research models leads to conservative results. An example of this is in the conceptual model for radionuclide retardation in the buffer where, in the Reference Case (i.e. Base Scenario plus the Reference dataset), potential nuclide sorption on the overpack corrosion products is conservatively neglected and only evaluated as an alternative "Model uncertainty case" for the Base Scenario.

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24 A research model may be used to simulate detailed processes, for example, corrosion of the overpack to assess how it is affected by various factors. However, in the SA, simplified models are used which do not have this level of detail. For the example of overpack corrosion, only failure time or a range of times (to reflect uncertainty) may be taken over into the SA

- *Parameter uncertainty* – where the intrinsic variability of physical parameters used in mathematical models is dealt with by assigning distributions or ranges to the parameters. Alternative calculation cases using a range of values for a parameter illustrate the impact of uncertainty for this parameter for a particular scenario. In practice, a value judged to be "realistic" and a conservative value, which leads to greater consequences, are often selected to keep the number of calculation cases to a manageable number.

From the H12 assessment context, there is no requirement to provide statistical distributions of the assessment end-point (i.e. annual individual dose) and so a full probabilistic implementation of the assessment models is not presented. Modelling results are consequently in a very similar format to that found in Nagra (1994a; 1994b) and this is in marked contrast to AECL EIS or YMP where results are expressed in probabilistic terms[25].

As noted in NEA (1997), there can be considerable overlap between the different forms of uncertainty in safety assessments. This is well demonstrated in H12 by the way in which the alternative conceptualisations of the geosphere are used to derive the most appropriate simplified representation of the fracture system. A superposition of one-dimensional fractures or "pathways", in which each pathway represents a set of geosphere fractures with similar transmissivities (the one-dimensional multi-pathway model is described in section 2.4 and Fig. 2.4.2), is used in the SA rather than a full 3-D fracture network. The justification for the use of the simpler model is numerical experiments which demonstrated that the multi-pathway model conservatively approximates transport in the stochastically generated 3-D fracture network. Thus inherent variability in the natural system is used as input to the assessment model. Similar approaches are found in Nagra (1994b).

In other parts of the model chain, conceptual uncertainty is more clearly evident in its own right. For example, the generic nature of the assessment requires that attention be given to a variety of potential geosphere-biosphere interfaces, ranging from coastal releases to mountain and hillside releases. This broad range of possibilities, arising as it does from the wide scope implied by a generic assessment, is not found in other assessments where the regional context is more clearly specified.

25 It should be noted that the fully probabilistic approach has been criticised in reviews due to the difficulty of presenting results to non-specialist audiences. (This was particularly a problem for the complementary cumulative distribution functions (CCDF) used in the YMP programme.) The approach has also been criticised for the great difficulty in determining where in the modal chain a particular parameter (e.g. reduced matrix diffusion) affects this calculated dose

The approach to assessment in Canada (AECL, 1994), where the emphasis is on a full statistical evaluation of radionuclide releases, contrasts markedly with H12. In H12, as in the Swiss approach, the emphasis is on the characterisation of uncertainty by systematic variation. The identification of the Base Case and the calculational Reference Case is therefore of paramount importance as a benchmark against which other calculation cases are compared.

There have been a number of advances in scenario analysis in recent years. Formal methods discussed by NEA (1997) are employed – interaction matrices (IMs) as well as Process Influence Diagrams (PIDs). These methods lead to the identification of alternative conceptual representations as well as allowing for the possibility of alternative evolution scenarios. Within each of the broadly defined scenarios, systematic parameter variations are employed to quantify model uncertainty as well as parameter uncertainty. This approach, whereby individual parameters and groups of parameters are set to credible extreme values from parameter ranges, provides information on the response of the system to parameter uncertainty without the need to carry out a full probabilistic run with detailed statistical analysis. This is comparable to the Swiss approach (Nagra 1994a; 1994b) and clearly differs from the AECL EIS, YMP and WIPP methodology and somewhat from that employed in SR 97[26] and TILA 99.

Given the extremely broad range of scenarios to be considered, arising from the generic site context for H12, this is a reasonable and practical approach. Full probabilistic safety assessment is acknowledged to be extremely parameter intensive in that the derivation of full probability density functions (pdfs) for model parameters can be very time consuming (NEA, 1997). For the generic nature and stated aims of the H12 study, it is right to concentrate on the identification and characterisation of conceptual uncertainty. Furthermore, the greater transparency and ease of communication of results from a deterministic assessment makes it preferable for presentation to non-specialist audiences.

4.5 Comparison of the treatment of near-field features and processes

In its treatment of near-field features and processes, H12 aims to use realistic assumptions regarding models and data, where these are well supported by experimental evidence, and conservative assumptions elsewhere. The construction of conservative/realistic models draws on the "site-generic"[27] R&D conducted after H3. This section aims to place in perspective the H12 treatment of:

- Waste-form dissolution
- Overpack corrosion and failure mechanisms
- Bentonite porewater chemistry
- Near-field solubility, sorption and diffusivity of radionuclides

26 SR 97 states that where probabilistic analyses are required to arrive at a measure of risk (for comparison with repository acceptance criteria), a statistical distribution for a parameter will only be used where there is some statistical material on which to base it. Otherwise, a defined "reasonable" value and a "pessimistic" value will be used

27 "Site-generic" implies that data is collected from a single site (or limited number of sites) with the intention to produce a coherent generic dataset (which may not result from an assemblage of data from many different sites) rather than to represent a specific site, as would be the case with "site-specific" data. In H12, a more generic approach is taken in which, although emphasis is placed on data from particular in-situ experiment sites (mainly Tono and Kamaishi), these are supplemented by relevant literature data

by means of comparison with other assessments that adopt comparable EBS design options.

4.5.1 Overpack corrosion and mechanical failure

The similarity of designs for the carbon steel reference overpack in H12, H3 and Kristallin-I would be expected to result in similar processes being identified for corrosion and eventual mechanical failure. In all three assessments, the overpack is designed to remain unbreached for at least 1000 years. Following failure, the overpack is conservatively assumed to offer no resistance to water ingress or radionuclide egress; credit is taken, however, for the redox buffering effects of the overpack corrosion products.

General corrosion processes:

Table 4.5.1 gives the general corrosion processes that were identified in the three assessments and the extent of corrosion estimated to arise from each process.

Rates for individual processes differ a little between the assessments. For example, H12 adopts a conservative assumption of highly non-uniform reaction (pitting) with trapped oxygen when calculating the maximum extent of this process. Combining the processes, however, a similar maximum extent of corrosion is estimated and a similar corrosion allowance is built into the overpack design.

Tab. 4.5.1: General corrosion processes and depth of corrosion (including pitting) over 1000 years

Corrosion process	H12	H3	Kristallin-I
Reaction with trapped oxygen, enclosed at the time of emplacement	1.8 mm (average) 11.8 mm (maximum)	10 mm	< 1 mm
Anaerobic reaction with water	20 mm	20 mm	20 mm (maximum)
Bacterial corrosion	2 mm	2 mm	9 mm (assumes all sulphate reaching overpack is reduced to sulphide)
Combined processes (maximum extent)	33.8 mm	32 mm	< 30 mm
Corrosion allowance	40 mm	50 mm	50 mm

Localised corrosion of steel:

In both H12 and Kristallin-I, it is argued that, as long as the overpack provides sufficient radiation shielding to ensure that the production of radiolytic oxidants is negligible (supported by laboratory data), then localised corrosion due to spatial separation of cathodic and anodic partial reactions will not occur (Section 4.1.1.3.2 in H12 Supporting Report 2). Nevertheless, in H12 pitting is assumed to occur and included as an empirical pitting factor. Kristallin-I discusses the possibility that stress corrosion cracking (SCC) could provide more rapid corrosion in areas with high residual stresses (e.g. due to welding). This could, however, be avoided either by heat treatment after welding, or by bolting or screwing the overpack lid.

Design and mechanical strength:

The overpacks are designed to:

- Withstand the external hydrostatic pressure and additional forces due to buffer swelling, corrosion volume expansion and rock deformation for at least 1000 years
- Provide radiation shielding at the overpack surface for the operational period.

In addition, the maintenance of a wall thickness of at least 150 mm over a 1000 year period is prescribed in H12 in order to avoid significant production of radiolytic oxidants that might contribute to corrosion.

Overpack wall thicknesses and the design external isostatic pressures are shown in Table 4.5.2.

Tab. 4.5.2: Comparison of overpack wall thicknesses and the design external isostatic pressures

	H12	**H3**	**Kristallin-I**
Design external isostatic pressures [MPa]	10.7 (hard rock; 1000 m depth) 6.8 (soft rock; 500 m depth)	55	30
Corrosion allowance [mm]	40	50	50
Total wall thickness (cylindrical shell), minus corrosion allowance [mm]	150	250	200

The wall thicknesses considered in the different assessments are similar, although Kristallin-I and H3 assume higher design external isostatic pressures than H12. In the case of H3, this is due to some very conservative assumptions and, in the case of Kristallin-I, it results mainly from a higher estimated bentonite swelling pressure than that of H12[28]. In H12, the wall thickness is, in fact, determined by a combination of the corrosion allowance and the need to provide radiation shielding to avoid radiolysis. The latter requirement gives more than adequate mechanical strength (see Table 4.6-4 in H12 Project Overview Report), particularly as, in all cases, the mechanical failure analysis is based on that for industrial pressure vessels which is intentionally very conservative.

4.5.2 Waste-form dissolution

In addition to H12, examples of safety assessments that consider a borosilicate glass waste form are H3 and Kristallin-I. In all three assessments, once water contacts the glass surfaces, it is assumed that radionuclides are released congruently with waste-form dissolution. Critical parameters determining the rate of release are the dissolution rate (per unit area of glass) and the surface area of the glass. The dissolution rate and initial surface area of the glass[29] from each assessment are compared in Table 4.5.3, along with the lifetime of a glass block.

Tab. 4.5.3: Comparison of dissolution rates and initial surface areas of the glass

	H12	H3	Kristallin-I
Dissolution rate	3.7×10^{-4} kg m^{-2} a^{-1} (10^{-3} g m^{-2} day^{-1})	5.2×10^{-2} kg m^{-2} a^{-1} (1.4×10^{-1} g m^{-2} day^{-1}) for fresh-reducing-high pH groundwater	3.8×10^{-4} kg m^{-2} a^{-1}
Ratio of initial surface area of fractured block to that of intact block	10	10	12.5
Glass block lifetime	7 x 10^4 years	4.5×10^2 years	~ 10^5 years

28 The swelling pressure of bentonite is highly sensitive to its density, as shown in Figure 4.6-6 in H12 Project Overview Report. The dry density of the bentonite used in the buffer is 1.6 Mg m^{-3} in H12 (bentonite/sand mixture, ratio 70:30), 1.7 Mg m^{-3} in Kristallin-I and 1.8 Mg m^{-3} in H3

29 The initial surface area will be increased from the geometric surface area due to fracturing during cooling and corrosion expansion of the overpack

The dissolution rates used in H12 and Kristallin-I, which are based on long-term experiments, assume "saturation" of the water[30] at the glass surface with dissolved silica and are similar in both assessments. The assumption of silica saturation is justified by the expected slow rate of transport of silica into the surrounding buffer and the fact that the bentonite porewater is saturated with silica. H3 took a more conservative approach, and estimated the silica concentration at the glass surface using a steady-state model for transport of silica through the buffer and assuming that the bentonite porewater silica concentration was zero. This resulted in a higher glass dissolution rate.

The assumed ratio of initial surface area of fractured block to that of intact block was similar for all three assessments.

4.5.3 Bentonite porewater chemistry

The composition of the porewater in the bentonite buffer is usually considered to be groundwater which has been modified by reaction with the bentonite minerals. The dominant constituent of the bentonite is usually Na-montmorillonite, which can undergo various ion exchange reactions. Reactions with the impurities, such as calcite, silica minerals, pyrite etc. depending on the specific bentonite composition, can also be an important part of the modification of the groundwater. Some safety assessments also include redox reactions with the products of the overpack corrosion. The Kristallin-I and H3 models for bentonite porewater form the focus of the comparison with H12 as, in these SAs, the bentonite porewater composition is explicitly defined. In other SA documentation, the bentonite porewater is not given although there may be some discussion of the models used to produce it. For example, SR 97 uses the Wanner model for bentonite porewater composition (Wanner 1986) to produce a single non-site specific water composition for deriving radionuclide solubilities and Kds but no composition is specified in the SA report or accompanying data compilation (SKB 1999c).

The chemical composition of the bentonite porewater in H12 was derived by a thermodynamic model, which in turn was based on experimental data produced by JNC. The approach used is termed an "empirical equilibrium model" and includes a number of solution, surface and mineral reactions. It is acknowledged in H12 that a full chemical understanding of highly compacted bentonite systems is still lacking. The approach focuses, however, more than earlier safety assessments, on the state-of-the-art mechanistic understanding gained from low solid-to-liquid ratio experiments. The processes included are given in Table 4.5.4, where also model assumptions, features, considered processes and main results (pH, Eh) are given. These are compared to those of H3 and Kristallin-I.

30 Glass is metastable, hence saturation in the sense applied to dissolution of stable crystalline solids does not occur. Models/empirical data show that the initial high glass dissolution rates decline to a slow "long-term rate" as Si reaches a fixed value

Tab.4.5.4: Comparison of assumptions, features and processes used for deriving bentonite porewater

	H12	**H3** [1]	**Kristallin-I**
Ion exchange included	yes	yes	yes
Protonation/deprotonation at edge surface	yes	no	no
Calcite equilibrium	yes	yes	yes
SiO_2 equilibrium	chalcedony	quartz	quartz
Other silicate equilibria (e.g. kaolinite)	no	no	yes
Magnetite equilibrium	yes	yes	yes
Pyrite oxidation	yes	yes	no
Type of bentonite considered	Kunigel-VI	Kunigel-V1	MX-80
Assumed temperature	60 °C	25 °C	50 °C
Thermodynamic data base	JNC-TDB[2]	PHREEQE/ NEA/PNC-TDB [3]	Nagra-TDB[4]
Thermodynamic calculational code used	PHREEQC[5]	PHREEQE[6]	MINEQL[7]
Sensitivity analysis	yes	no	no
Number of reference waters considered	1[9]	4	2
Time evolution modelled	Yes	yes	no
pH Reference Case groundwater [8]	8.5	8.7	8.5 - 9.0
Eh Reference Case groundwater (mV)[8]	-281	-283	-367 to -396

Notes
1: Fresh, reducing, high pH used as the Reference Case groundwater composition. Three other water compositions were considered as alternatives.
2: Yui et al., 1999
3: Parkhurst et al., 1980; Müller, 1985; Yui et al., 1992
4: Pearson and Berner, 1991; Pearson et al., 1992
5: Parkhurst et al., 1995
6: Parkhurst et al., 1980
7: Schweingruber, 1982
8: pH and Eh of Reference groundwater
9: For alternative cases, other groundwaters (saline, high pH) were also considered.

Surface reactions:

All approaches include ion exchange reactions. Unlike previous SAs, protonation/deprotonation reactions at clay edge sites were considered in H12. These were based on interpretation of recent experiments performed by JNC.

Mineral equilibria:

The approaches used in the three assessments are similar. H3 and Kristallin-I did not consider dissolution of smectite and/or other aluminium silicates and the significant difference in porewater chemistry between H12 and H2 due to the protonation / deprotonation reactions mentioned above.

Tab. 4.5.5: Comparison of calculated bentonite porewater compositions (Note differences in calculation temperatures - Tab. 4.5.4)

	H12	H3 (1)	Kristallin-I	
			Low salinity	High salinity
pH	8.4	10.3	8.97	8.49
Eh (mV)	-276	-432	-396	-367
Total elemental concentrations (mol dm^{-3})				
Na	2.8×10^{-3}	4.4×10^{-2}	7.8×10^{-2}	2.4×10^{-1}
Ca	5.3×10^{-5}	~0	1.1×10^{-5}	5.5×10^{-5}
K	1.2×10^{-4}	1.1×10^{-4}	1.7×10^{-4}	5.7×10^{-4}
Mg	4.2×10^{-6}	3.0×10^{-5}	6.7×10^{-5}	4.8×10^{-4}
Fe	2.0×10^{-9}	4.2×10^{-12}	n.c.	n.c.
Al	3.4×10^{-7}	n.c.	1.0×10^{-4}	3.6×10^{-5}
C	1.6×10^{-2} (2)	2.5×10^{-2}	5.2×10^{-2}	1.8×10^{-2}
S	1.1×10^{-4} (3)	1.2×10^{-4}	4.5×10^{-3}	1.6×10^{-2}
B	2.9×10^{-4}	n.c.	n.c..	n.c.
P	2.9×10^{-6}	n.c.	n.c..	n.c.
F	5.4×10^{-5}	n.c.	7.0×10^{-4}	1.9×10^{-4}
N	2.3×10^{-5}	n.c.	n.c..	n.c.
Cl	1.5×10^{-5}	n.c.	3.0×10^{-3}	1.9×10^{-1}
Si	3.4×10^{-4}	7.8×10^{-4}	2.6×10^{-4}	1.9×10^{-4}

Notes:
1: Based on "fresh, reducing, high pH"-type groundwater
2: Inorganic C = 1.6×10^{-2} mol dm^{-3}; $CH_{4\,(aq)}$ = 8.1×10^{-10} mol dm^{-3}
3: S(VI) = 1.1×10^{-4} mol dm^{-3}; S(-II) = 3.8×10^{-9} mol dm^{-3}
n.c.: not calculated

Calculation procedure:

Only one reference water is defined in H12 (unlike H3 and Kristallin-I). This is complemented by the alternative calculation cases in which the SRHP groundwater (saline, reducing, high-pH) is considered. The calculations are performed with an in-house thermodynamic database and the geochemical calculational code PHREEQC. A sensitivity analysis in H12 includes computations with another database and with another clay material (Wyoming bentonite MX-80). The salt impurities in the bentonite are omitted because it is assumed that these will have been dissolved and transported from the EBS at the time of overpack failure (> 1000 years).

The porewater evolution is treated in a similar way in H3 and H12, using a mixing tank model (see, for example, section 5.3.1.2.3 of H12 Supporting Report 3).

Results:

The calculated bentonite porewater compositions are given in Table 4.5.5. Given that there were significant differences in starting groundwater composition, these have resulted in corresponding differences in the bentonite porewater concentration for some elements, despite the bentonite-water reactions. Na and Cl are good examples of this. Other element concentrations reflect the interactions, for example Ca and C, influenced by calcite dissolution/equilibrium, are very consistent between SAs. (The reason for a value of 0 mol dm^{-3} (10^{-6} mol dm^{-3}) for Ca in H3 is that Ca concentration of the groundwater is very low, of the order of 10^{-5} mol dm^{-3}, and Ca from the dissolution of calcite is taken up by ion exchange onto the Na-montmorillonite.) Likewise, Si and K tend to reflect reaction with chalcedony and microcline rather than simply groundwater composition.

4.5.4 Near-field solubilities, sorption and diffusivities

Solubilities:

In H12, radionuclide solubilities for the near field are derived based on input from the geochemical code PHREEQE and the JNC thermodynamic database (JNC-TDB – Yui et al. 1999). In cases where relevant experimental data indicate higher solubilities, these values were taken over. Co-precipitation effects are considered for some elements. In developing the JNC-TDB, emphasis was put on the traceability and the scientific reliability of the origin of the thermodynamic data.

The comparison in Table 4.5.6 shows that additional elements, such as Nb, Sm, Pb and Ac are considered in H12, compared to H3 and Kristallin-I. Another difference is that a single value for each element is used in contrast to a range of values as used in H3 to reflect a large uncertainty in the solubility limits for several elements (especially for the actinides). Two alternative values are used in Kristallin-I – a realistic value and a more conservative value. TILA 99 uses a set of 5 values for solubility of each element to reflect the influence of different groundwater composition (in particular saline and non-saline, oxidising and reducing conditions) as well as uncertainty in values (conservative and very conservative). SR 97 uses site-specific solubilities for each of the three sites assessed: the wide range of groundwater composition for the sites, from fresh to saline, leads to considerable differences in the elemental solubilities used at each site.

Tab. 4.5.6: Comparison of Reference Case solubility limits (mol dm^{-3}) for 5 recent safety assessments

Element	H12	H3	Kristallin-I		TILA 99	SR 97
			realistic	*realistic-conservative*	*conservative non-saline*	*Ceberg fresh*
Se	3×10^{-9}	1×10^{-8} - 8×10^{-7}	1×10^{-8}	6×10^{-7}	1×10^{-6}	2.6×10^{-9}
Zr	1×10^{-6}	1×10^{-10} - 3×10^{-8}	5×10^{-9}	5×10^{-7}	5×10^{-8}	2.5×10^{-9}
Nb	1×10^{-4}	-	-	-	1×10^{-3}	1.4×10^{-3}
Tc	4×10^{-8}	1×10^{-12} - 4×10^{-8}	1×10^{-7}	High	5×10^{-8}	7.7×10^{-9}
Pd	1×10^{-9}	1×10^{-9} - 4×10^{-6}	1×10^{-11}	1×10^{-6}	1×10^{-8}	4.2×10^{-9}
Sn	5×10^{-6}	2×10^{-8}	1×10^{-5}	1×10^{-5}	5×10^{-6}	4.7×10^{-9}
Cs	High	-	High	High	-	-
Sm	2×10^{-7}	-	-	-	1×10^{-5}	8.0×10^{-7}
Pb	2×10^{-6}	-	-	-	-	-
Ra	1×10^{-12}	-	1×10^{-10}	1×10^{-10}	1×10^{-7}	1.2×10^{-4}
Ac	2×10^{-7}	-	-	-	-	-
Th	5×10^{-6}	2×10^{-10} - 6×10^{-4}	5×10^{-9}	1×10^{-7}	5×10^{-7}	1.2×10^{-9}
Pa	2×10^{-8}	-	1×10^{-10}	1×10^{-7}	1×10^{-8}	3.2×10^{-7}
U	8×10^{-9}	3×10^{-10} - 4×10^{-5}	1×10^{-7}	7×10^{-5}	3×10^{-7}	1.3×10^{-7}
Np	2×10^{-8}	2×10^{-12} - 5×10^{-9}	1×10^{-10}	1×10^{-8}	5×10^{-8}	5.8×10^{-8}
Pu	3×10^{-8}	3×10^{-11} - 3×10^{-2}	1×10^{-8}	1×10^{-6}	5×10^{-7}	1.4×10^{-10}
Am	2×10^{-7}	6×10^{-9} - 1×10^{-7}	1×10^{-5}	1×10^{-5}	5×10^{-7}	9.3×10^{-8}
Cm	2×10^{-7}	-	6×10^{-8}	1×10^{-5}	5×10^{-8}	9.0×10^{-10}

For the elements U, Ra and Se, a significantly lower value is proposed in H12 compared to the earlier H3 and Kristallin-I assessments. The value for U is also significantly lower than for the more recent TILA 99 (conservative, non-saline dataset) and SR 97 (Ceberg dataset), however the H12 Se value is very similar to that used in these assessments. The difference between the H12 and SR 97 solubility for Ra is eight orders of magnitude, although this reflects the conservatism of the SR 97 value, which is very significantly higher than for other assessments. The solubility limit given for Th is higher in H12 than in the other assessments, due to consideration of carbonate complexes.

Sorption and diffusivity:

From the outset, it should be noted that there is considerable academic controversy associated with the quantification of solute transport through microporous media such as highly compacted bentonite. The fact that the nanometre-scale pores have charged surfaces influences the structure (and hence activity) of contained water causing the fundamental assumptions of most chemical thermodynamic or mechanistic sorption models to break down. This, of course, limits the reliability of the models and empirical databases used to derive bentonite porewater chemistry (see previous section). However, as such chemistry is not used directly in the SA model chain, this is not so critical.

The data and models for radionuclide uptake onto, or transport through, compacted bentonite are clearly critical to evaluation of its barrier role. Although the key experiments directly measuring radionuclide diffusion at relevant compaction levels give a clear picture of the relative mobility of different species, these can be interpreted by very different models – for example:

- Considering different porosity accessible to ions of different charge or size
- Different diffusion processes occurring simultaneously (e.g. in solution and "surface diffusion")
- Differential mobility of various solution species of a single element
- Non-linear sorption, etc.

The way in which detailed research models are simplified for SA purposes is variable.

The transport of radionuclides through bentonite is treated in a similar fashion in all the SA studies. Diffusion is considered to be the dominant transport mode, with element-specific apparent diffusion coefficients (D_a) determined either directly from diffusion experiments or calculated from the distribution coefficient (Kd) via the equation:

$D_a = D_e/(\varepsilon + \rho \, Kd)$,

where D_e is the effective diffusion coefficient, ε and ρ are the porosity and density of the bentonite, respectively. H12 considers a 70:30 bentonite-sand mixture; most other studies deal with pure bentonite. Relevant values of the transport parameters for a number of SA studies, including specific data for several key radionuclides, are compared in Table 4.5.7.

There are relatively few data for D_e in bentonite, and thus, in some assessments (e.g. H3, SITE 94 and Kristallin-I), a single value is selected for all species, based on measurements with a conservative tracer such as HTO or I^-. Other studies assume significant differences in D_e for different types of species (e.g., SKB 91 and TILA 99). These are based on some measurements that suggest that surface diffusion may increase D_e values for some species (e.g. Cs), and that anion exclusion may result in low D_e values for others (e.g. I and Cl in SKB 91 and TILA 99). In H12, data from JNC's own laboratory are used to show that D_e values for Cs, Se and HTO are unaffected by the presence of silica sand at a level of 30 %.

Distribution coefficients can be determined from batch studies with uncompacted bentonite, or can be calculated from measured D_a values. There can be significant differences between values determined with these two approaches, with batch studies often giving the higher value. In the case of H12, values derived from measurements of D_a are used.

It can be seen from Table 4.5.7 that, despite different mechanistic assumptions, there is reasonable consistency in D_a values for a given radionuclide in the various assessments. Apart from Cs, the greatest variations occur in the case of nuclides that are redox-sensitive, e.g. Tc and Se, as this affects sorption properties via the assumed speciation (anionic vs. cationic form).

The impact of variations in parameter values for mass transport through bentonite on release to the far field is strongly affected by the boundary conditions, which vary considerably for the various assessments. In H3, a zero-concentration outer boundary condition at the bentonite / rock interface was used[31], whereas in H12, TILA 99 and Kristallin-I, the diffusive flux through the bentonite was matched to the product of the radionuclide concentration and the total groundwater flow through the EDZ. The SKB 91 model used a resistor network, whereas, in SITE 94, the near-field rock was modelled as a porous medium, with the flux determined by an interfacial diffusion coefficient and the outer boundary of the near-field rock having a zero-concentration boundary condition.

4.6 Comparison of the treatment of geosphere features and processes

In its treatment of geosphere features and processes, H12 aims at a conservative, but plausible, representation of a range of potential host-rock types, even though, at the current stage of Japanese repository planning, no site-specific data are available. This is considered important if H12 is to provide useful input for future siting and site-characterisation studies. The present section aims to place in perspective the H12 treatment of:

- Groundwater flow
- Matrix diffusion
- Geosphere sorption

by means of comparison with other assessments. Attention is focused on silicic crystalline rock (e.g. granite), which is the reference host rock type for H12.

31 The influence of this assumption on the results for H3, compared to Kristallin-I was examined in some detail in Neall (1994)

Tab. 4.5.7: Comparison of Reference Case parameters critical to the evaluation of radionuclide diffusion through bentonite in various safety assessment studies

	H12	H3	SITE 94 conservative case	SKB 91	TILA 99 conservative case	Kristallin-I conservative case
Bentonite porosity	0.41	0.33	0.36	0.05 (anions) 0.25 (others)	0.05 (anions) 0.43 (others)	0.38
Effective diffusion coefficient D_e [m^2 a^{-1}]x10^{-2}]	Cs 2 Se 0.6 Others 1	1	0.1	C, Se 0.3 Tc(R)[1] 0.3 I,Cl 0.008 Tc(O)[2] 0.008 Sr, Cs 8	N[3] 0.3 A[4] 0.02 C[5] 20 (different values for saline and non-saline conditions)	0.63
Bentonite dry density, [x 10^3 kg m^{-3}]	1.6	1.8	1.7	1.75	1.6	1.7
Apparent diffusion coefficient D_a [m^2 a^{-1}] Cs Se U Pu Tc I	 1×10^{-3} 2×10^{-2} 6×10^{-6} 6×10^{-7} 6×10^{-5} 3×10^{-3}	 5×10^{-4} 4×10^{-3} 5×10^{-5} 5×10^{-7} 4×10^{-3} -	 6×10^{-5} 3×10^{-4} 6×10^{-6} 6×10^{-7} 3×10^{-3} 3×10^{-3}	 9×10^{-3} 6×10^{-4} - 4×10^{-8} 2×10^{-5}(R) 2×10^{-3}	 6×10^{-4} 3×10^{-3} 3×10^{-5} 6×10^{-6} 2×10^{-4} 3×10^{-3}	 3×10^{-3} 3×10^{-3} 7×10^{-6} 7×10^{-6} 7×10^{-5} -
Distribution coefficient Kd [m^3 kg^{-1}] Cs Se U Pu Tc I	 0.01 0 1 10 0.1(R)[1] -	 0.001 0.001 0.1 10 0.001 -	 0.01 0.002 0.1 1 0(O)[2] 0	 0.05 0.003 - 50 0(O)[2] 0	 0.04 0 0.05 0.3 0(O)[2] 0	 0.001 0.001 0.5 0.5 0.05 -

Notes:
1 Reduced
2 Oxidised
3 Neutral species
4 Anions
5 Cations

4.6.1 Groundwater flow

An understanding of groundwater flow is required in order to evaluate the performance and longevity of the EBS, and to evaluate the performance of the geosphere transport barrier.

Groundwater flow in fractured media, as considered in H12, depends critically upon:

- The hydraulic gradient
- The transmissivity distribution of water-conducting features
- The large-scale heterogeneity of the rock – in particular, the distribution and spatial density of fractures or other water-conducting features
- The small-scale heterogeneity of these features and, in particular, the concentration of flow in discrete channels.

In addition, in order to evaluate the performance of the geosphere transport barrier, a conceptual model of the transport path or paths from the repository to the biosphere is required. For example,

- H12 considers a set of transport paths, each comprising a horizontal section through the host rock around the repository, and an upwardly-directed section through a fault to the overlying sediments
- H3 considers a transport path comprising a single fracture or a continuous porous medium linking the EBS to a well or a river from which drinking water is extracted
- TILA 99, SITE 94 and SKB 91 consider ranges of transport paths in regional-to-site scale flow analyses of groundwater flow, and then obtain representative transport parameters from the results
- Kristallin-I considers a transport path comprising a single, upwardly-directed section from the repository to the overlying, higher-permeability crystalline basement (or equivalently, a horizontal section through the host rock around the repository to a major water-conducting fault, the barrier function of which is neglected in the Reference Case).

Critical groundwater-flow parameters used in these selected safety assessments are compared in Table 4.6.1.

It is pointed out in TILA 99 that, for evaluation of the performance of the geosphere transport barrier, it is the lumped parameter $cL/(Ti)$ (the transport resistance)[32] that described the main contribution of the hydraulic properties of the rock, and it is this parameter that is presented in the TILA 99 report. The larger this parameter, the more effective the geosphere transport barrier is expected to be.

32 See Table 4.6.1 for notation. In the notation of TILA 99, the lumped parameter is WL/Q. In the notation of SKB 91, it is $t_w a_R/\varepsilon_f$. Radionuclide transport times in the geosphere depend not only on this parameter, but also on matrix diffusion and sorption, which are discussed in sections 4.6.2 and 4.6.3, respectively

Tab. 4.6.1: Comparison of parameters critical to the evaluation of groundwater flow (Reference Case dataset unless otherwise specified)

	H12 (host rock)	H12 (fault)	H3 (single fracture)	TILA 99	K-I	SITE 94	SKB 91 (central values)
Hydraulic gradient, i	0.01	0.01	0.05	-	0.02	-	-
Trans-missivity distribution T [m^2 s^{-1}]	log-normal mean =-9.99 st. dev. = 1.07	10^{-7}	$10^{-10} \times f$	-	log-normal mean =-9.24 st. dev. = 0.4	-	-
Fracture density, f [m^2 per m^3 of rock]	0.8	-	10	-	0.04	-	-
Channelling factor[1], c	0.5	0.5	1	-	0.06	0.04 (assuming a 2.5 m fracture spacing)	-
Migration distance, L [m]	100	800	10, 100, 1000	-	200	500	-
"Transport resistance"[2] cL/Ti [s m^{-1}] (a m^{-1})	5×10^{13} (1.5×10^{6})	4×10^{11} (1.3×10^{4})	2×10^{12} for L = 100 m (6.3×10^{4})	1.6×10^{12} (5×10^{4})	1.0×10^{12} (3.3×10^{4})	2.1×10^{12} ($6.7\text{x}10^{4}$)	2.2×10^{13} (7.0×10^{5})

Notes:
1: Proportion of fracture surface in contact with flowing water
2: For H12 (host rock), and Kristallin-I, this is evaluated in the above table using the mean of the log-normal transmissivity distribution.

The transport resistance provided by the host rocks are similar in H12 (for fractures with the geometric mean transmissivity) and SKB 91 (central values), and about an order of magnitude higher than the transport resistance in H3, TILA 99, Kristallin-I, and SITE 94. This difference may be attributed, at least in part, to assumptions regarding small-scale heterogeneity and, specifically, the "channelling factor", i.e. the proportion of fracture surfaces taken to be in contact with flowing water. This is an order of magnitude higher in H12, compared to Kristallin-I and SITE 94.

The same basic fracture flow model with some changes to the parameters used was applied in H12 to the low permeability, sediments also considered as host rocks. This procedure was justified by the observed presence of fractures in hard sediments but it will be important to consider other structures in such rocks (e.g. sand channels in siltstones, flow beds in tuffs etc.) which could have very different solute transport properties (c.f. Nagra 1989).

4.6.2 Matrix diffusion

Matrix diffusion, in addition to sorption, can provide an efficient retardation mechanism for nuclides migrating through the geosphere.

The retardation effect of matrix diffusion depends critically upon:

- The effective diffusion coefficient (the product of a pore diffusion coefficient and the matrix porosity)
- The extent of connected porosity adjacent to a fracture, which defines, in part, the maximum depth to which diffusion occurs
- (also sorption on matrix pore surfaces for sorbing nuclides, see section 4.6.3).

Matrix diffusion parameters used in selected safety assessments are compared in Table 4.6.2 (sorption on matrix pore surfaces is discussed in Section 4.6.3).

The TILA 99, SITE 94 and SKB 91 assessments adopt porosities and effective diffusion coefficients that are significantly smaller than those of H12, H3 and Kristallin-I. TILA 99 is the only one of the assessments explicitly to model the higher-porosity matrix near the fracture, as well as lower-porosity matrix at greater distances. Such heterogeneity is discussed at length in H12 and Kristallin-I, but, in the transport model, single values are conservatively assigned to these parameters. TILA 99 is, furthermore, the only one of the assessments to assign different properties to the matrix according to whether the migration of anions or non-anions is being considered. Again, other assessments take anion exclusion into account by setting conservative parameter values that are applicable to all nuclides.

Tab. 4.6.2: Comparison of parameters critical to the evaluation of matrix diffusion

	Effective diffusion coefficient [$m^2\ s^{-1}$]	**Porosity [%]**	**Maximum diffusion depth [m] [1]**
H12 (host rock and fault)	3×10^{-12}	2	0.1
H3	10^{-14} to 10^{-12}	1	0.1
TILA 99 (non-saline water) 2-layer matrix	10^{-14} for anions 10^{-13} otherwise	0.1 for anions 0.5 otherwise	0 – 0.01 m from fracture
	10^{-15} for anions 10^{-14} otherwise	0.02 for anions 0.1 otherwise	0.01 – 0.1 m from fracture
Kristallin-I	1.5×10^{-12}	5	0.05
SITE 94	3×10^{-14}	0.1	0.05
SKB 91	1×10^{-13}	0.5	*unlimited*

Notes:
1: Mechanistically, the degree of pore connectivity is what determines the extent of matrix diffusion. For simplicity in SA, modellers effectively express this by varying diffusion depth.

All these analyses and their supporting databases are based on crystalline host rock. The extent of matrix diffusion from fractures in sediments is not explicitly addressed in any SA study to date[33]. This is partly due to lack of appropriate data (see discussion in Mazurek et al. 1996 and references therein).

4.6.3 Geosphere sorption

H12, like other assessments, uses distribution coefficients (Kd) for individual radionuclides to represent the various processes that are lumped together under the term "geosphere sorption". Many batch experimental Kd data exist in the published literature for the radionuclides of interest but there are fewer data on intact rock samples[34]. Use of these experimental data assumes that sorption is independent of concentration (i.e. linear), reversible and that the Kd determined from (mainly batch) experiments can be extrapolated to the physical and chemical conditions pertaining in the geosphere. These assumptions are addressed in a variety of ways:

- By choice of experimental data (e.g. SITE 94, H3, H12, Kristallin-I, TILA 99 and others), so that only data for appropriate rock type(s), water composition (e.g. pH, saline/non-saline) and redox conditions are included in databases. In H3, the data used were mainly from the literature; in H12, the emphasis was on in-house data, augmented by literature values
- By use of explicitly conservative values (or an additional set of conservative values which takes into account possible perturbations from the reference conditions, for which a realistic-conservative data set are used e.g. Kristallin-I and TILA 99)
- By use of a non-linear isotherm where possible (e.g. Kristallin-I for Cs only)
- By comparison of results from the Kd calculations with those using mechanistic sorption models (e.g. SITE 94 for Np).

In fact, most assessments explicitly note that the difference in sorption behaviour of the safety-relevant nuclides for most rock types is much less important than the influence of different water compositions or, for some radionuclides, redox conditions. Even in Kristallin-I, where single mineral distribution coefficients are used with reference mineralogies to calculate bulk Kds for infill, unaltered and altered wallrock, rock type differences are too small to necessitate the use of separate Kd values. Thus sets of Kd values for radionuclides in different water compositions under oxidising or reducing conditions are compiled and rock composition is taken into account by the choice of the lower values where a range is found for different rock types.

A comparison of Kds for selected radionuclides for low salinity water under reducing conditions (Table 4.6.3) shows that variations of more than 2 orders of magnitude occur, in part reflecting differences in water composition, particularly pH, and specific redox

33 A maximum matrix diffusion depth of 0.1 m is used in the (low and intermediate level waste) repository SA of Valanginian Marl at Wellenberg (Nagra 1994a) but the derivation of this value is not clear

34 The majority of Kd data are produced in so-called "batch" sorption experiments where crushed rock and relatively large volumes of liquid (groundwater) are mixed and then spiked with radionuclides. These systems are not representative of in-situ conditions as the surface area of the crushed rock is much larger than in-situ (in a fracture, for example) and the rock/water ratio is also inappropriate. Intact rock samples are rarely used due to uncertainties about the representativeness of the sample surface (when compared to a fracture surface, for example). A more appropriate system is infiltration of radionuclide-spiked water through an intact rock core, but such systems are expensive and time-consuming, measure Rd rather than Kd and the parameters produced are model-dependent (unlike batch sorption Kd values)

conditions. Differences also arise from the different approaches to uncertainty in chemical conditions (e.g. Kristallin-I and TILA 99, as discussed above).

Overall, the values used in H12 are similar to those of Kristallin-I. The values used in H3 are characterised by the large ranges used to take uncertainty into account, but are generally significantly lower than the revised values used in H12.

H12, like TILA 99, Kristallin-I and SITE 94, conservatively neglects the effect of sorption on the surfaces and infill of host rock fractures, taking account only of sorption in the rock matrix. It was noted in H12 (Supporting Report 1), however, that the fracture infill did not have retardation properties significantly different from those of the matrix. This contrasts with TILA 99 and SITE 94, where clay- or iron-oxide-rich fracture infill material would be expected to have a significantly higher retardation potential than the matrix.

Tab. 4.6.3: Comparison of distribution coefficients (Kd [m^3 kg^{-1}]) for a selection of elements from various recent safety assessments. (Reference Case unless otherwise stated)

Element	H12[1]	H3[2]	Kristallin-I[3]	SITE 94[4]	TILA 99[5]
Se	0.01	0.001 – 0.05	0.01	0.01	0.0005
Zr	0.1	1 – 5	1	4	0.4
Tc	1	0.0005 – 0.05	0.5; 0.05[6]	0.01	0.2
Sn	1	0.05 – 0.25	0.5	0.1	0.2
Cs	0.05	0.001 – 1	0.042[7]	0.1	0.1
Pa	1	-	1	0.5	0.2
U	1	0.00001 – 0.25	1	5	1

Notes:
1 For granitic hostrock with fresh, reducing, high pH (FRHP) groundwater
2 For fractured media (granite and basalt) in FRHP groundwater
3 Realistic-conservative values for reference groundwater (low salinity, reducing)
4 For far-field, reducing, non-saline groundwater
5 Realistic values for non-saline, reducing groundwaters
6 Area West; Area East (different redox conditions)
7 Based on a non-linear isotherm and the maximum concentration in solution of Cs along the geosphere transport path

4.7 Comparison of the treatment of biosphere features and processes

4.7.1 The H12 biosphere model in context

The biosphere model in the H12 assessment is markedly more complex than that applied in the H3 study, where a simple dose conversion was made after taking into account dilution of the geosphere releases in groundwater or river water used for drinking. The additional detail included in the biosphere parallels increasing capabilities in other aspects of safety assessment and is in line with the AEC Guidelines (AEC, 1997) which require a broad description of the geological (including near-surface) environment.

In contrast to the H3 assessment, H12 evaluates radiological impact in terms of annual individual dose, rather than a measure of radiological hazard. This requires that effort is made to assess not just dilution in the biosphere but also the potential for transport and for accumulations of radionuclides in the environment. Exposure routes by which human may come into contact with contaminated media must also be represented.

The H12 assessment deals with a wholly generic site, thus there is a great deal of uncertainty as to the relevant surface environment, and hence the appropriate type of geosphere-biosphere interface.

It is widely recognised that a reference biosphere approach is most suitable (see, for example, BIOMOVS 1993; BIOMOVS II 1996a; 1996b; BIOMASS 1998) when considering the long timescales over which discharge from the deep geosphere to the near surface environment is expected to occur.

BIOMASS (1998) defines the reference biospheres concept as

> *... the set of assumptions and hypotheses that is necessary in order to provide a consistent basis for calculations of the radiological impact arising from long term releases of repository derived radionuclides into the biosphere.*

This lays great emphasis on the identification, classification and discussion of assumptions used to derive the biosphere assessment model. H12 utilises the Reference Biosphere Methodology (RBM), the development of which ran in parallel with the H12 assessment[35], to allow for the thorough identification and documentation of biosphere-related modelling assumptions and relationships in a way that has not been possible in many comparable, recent assessments.

However, the link between the conceptual models of the biosphere and the mathematical representations of relevant biosphere FEPs is not well established in H12. It is for this reason that the mathematical description of the biosphere has proceeded along somewhat conservative lines, with a relatively simplified approach being adopted compared to, for example, the biosphere model employed in Kristallin-I.

One of the principal differences between H3 and H12 is seen in the classification and evaluation of different exposure pathways and example exposure groups. H3 evaluated the potential radiological effect of drinking water concentrations whereas H12 considers a broad range of food types - derived from terrestrial as well as marine sources. In part, this range arises as a result of the consideration of different geosphere-biosphere

35 It remains under development at time of writing. It was due for completion in late 2000

interfaces, but it should also be recognised that many more food types are included in the H12 assessment than has been common in other contemporary assessments.

4.7.2 Biosphere model definition and the RBM

Four stages are identified in the Reference Biosphere Methodology:

- Definition of the **assessment context** - the purpose of the assessment, the performance measure, timescales, repository type, site conditions, societal context
- Definition of the **biosphere system description** - identification of system components based on a review of FEP lists
- **Exposure group definition** - identification of relevant habits and behaviour of potentially exposed (usually human) populations
- **Model development** - the translation of the conceptual description of the system and exposure groups into a practical assessment tool. This stage includes the application of data to the assessment model.

4.7.3 The H12 model definition

4.7.3.1 Assessment context

The national regulatory background has a strong influence on the assessment context and hence on the form of the biosphere assessment model. The AEC Guidelines (AEC, 1997) direct model development to a constant biosphere representation, employing exposure groups with present-day lifestyles and with annual individual dose as the chosen performance indicator, assessed over multiple exposure pathways. The societal context is therefore that of the present day.

The purpose of the assessment is to demonstrate the validity of the deep disposal concept in Japan. This is broadly comparable with the context employed in other, earlier assessments (e.g. KBS-3, Project Gewähr, AECL EIS, ENRESA 97). The waste type is vitrified HLW, although this has little bearing on the treatment of the biosphere. It should be noted that safety relevant radionuclides in the biosphere are not always similar to those for NF and geosphere transport analyses which provide RN fluxes into the biosphere, and hence the source term (i.e. waste type and inventory) should be carefully considered for the biosphere assessment.

There are similarities with many recent assessments in that dose is to be used as the end-point of the assessment calculations. Only TILA 99 limits the assessment to a single exposure pathway. Present day biosphere conditions are assumed in many assessments, but most others acknowledge that there will be some modification of the near-surface environment over the very long timescales involved. In Kristallin-I, modifications were applied to climate conditions as a test of sensitivity, whereas more detailed biosphere evolution was considered in SKB 91, SR97 and SITE 94. The Swiss regulations (HSK/KSA, 1993) require that the assessment be carried out until the peak radiological impact has been reached. A similar approach is taken in H12, in accordance with the AEC Guidelines (AEC, 1997).

Although the assessment parameter to be evaluated is annual individual dose, the role of the biosphere model in H12 is primarily to evaluate the annual dose per unit release, i.e., a biosphere conversion factor (BCF[36]) relating release rate to radiological impact. This is in common with Swedish and Finnish usage of biosphere models (SKB 91, SR97, SITE 94 and TILA 99) and does not take account of the dynamics of the biosphere and the time taken for peak radiological impact to arise. The UK (NIREX 97) follows a similar approach, but the Swiss (Nagra, 1994a) and Canadian (AECL, 1994) approaches have a biosphere model as an integral part of the assessment model chain (near-field, geosphere, biosphere, dose). Biosphere conversion factors can be calculated with such a model structure, with the peak dose for each radionuclide being normalised to the input flux to the biosphere. The results for the BCFs are directly comparable because - for deep disposal at least - the time for the biosphere model to reach equilibrium is usually short compared to the timescale of the release from the repository and transport through the geosphere.

Site conditions are very important in determining timescales and concentrations in the biosphere. The generic nature of the H12 model reflects this in that, unlike all the other comparable assessments, the concept must embody nation-wide characteristics whereas the others may be more properly described as region specific.

4.7.3.2 The geosphere-biosphere interface

The importance of the geosphere-biosphere interface has been stressed many times (BIOMOVS, 1993; BIOMOVS II, 1996a) and, with the implementation of the RBM, its role in the determination of the assessment model is made more obvious. The generic nature of the H12 concept requires that a suitable geosphere-biosphere interface, characteristic of Japan be employed.

H12 has many points of similarity with Kristallin-I, including the modelling of the biosphere as "the place where doses arise". It may be argued that this places a conservative bias on the modelling since many beneficial effects are neglected (such as retardation and dispersion in near-surface aquifers) and the release from the geosphere is maintained at its most concentrated form on reaching "the biosphere" where dilution in the near-surface waters takes place. Effectively the near-surface environment acts as a "pipeline", discharging radionuclide bearing groundwaters to river water. This choice is made in the Reference Case since most current irrigation abstraction in Japan is from rivers.

This reflects conditions in Japan but most contemporary assessments assume a different form of geosphere-biosphere interface. In the Swedish, Finnish and Swiss assessments, wells are considered an important interface as they bypass many surface dilution and dispersion processes. They are also important sources of water for the types of community existing at present and (from historical records) in the past. NIREX 97 also considers well abstraction, but the same biosphere conversion factors are employed in the calculation of risk and dose as were derived for the *natural discharge* case, where radionuclide bearing groundwater discharges to soils and to surface water bodies (in the form of stream or rivers). In the Nirex assessment, detailed (geosphere) modelling of the aquifer is carried out to determine the concentration in the geosphere-biosphere groundwater flow. The most recent Swedish example places emphasis on the potential for groundwater discharge into a number of different locations and ecosystem types.

36 The term "flux to dose conversion factor" is used in H12

AECL (1994) implements lake, soil and well interfaces with groundwater concentrations determined by the geosphere codes.

A further point of comparison is the BIOMOVS II (1996b) study, which implemented the aquifer as a compartment of the biosphere system. This was found to lead to potentially significant feedback mechanisms and to accumulation in the aquifer. The Kristallin-I biosphere employed a similar model of near-surface hydrology.

Alternative instances of geosphere-biosphere interface are taken into account in variant cases in H12. These allow an investigation of different areas (small – large farm), topography (plain, hill, mountain), and discharge to the marine environment rather than the terrestrial environment. Discharge to riverbed sediment is also considered, as is the direct use of well water. A further option investigates the influence of sorption in the aquifer, conservatively neglected in other options.

Taken together, these variants on the H12 Reference Case include many features common to the other assessments. The breadth of the investigations reflects the generic features of assessment with respect to the biosphere and the geosphere.

4.7.3.3 Biosphere system description

Description of the biosphere system is very comprehensive in H12 since it employs techniques first discussed in BIOMOVS II (1996a) and recommended by BIOMASS (1998). There is extensive use of interaction matrices as a means of linking biosphere FEPs in context. Coupled with a two stage screening process, the identification system phase provides for a concise review of the conceptual biosphere system that is not found in other contemporary assessment documentation. This emphasises the benefits of the Reference Biospheres Methodology.

With respect to the databases used in the H12 biosphere model (e.g. BIOMOVS II 1996a, it should be noted that these tend to represent western (particularly northern European) diet and farming practices, which may not be entirely appropriate to the Japanese context. Further work may be needed to more realistically represent Japanese lifestyles in the data used.

5 THE FINDINGS OF SAFETY ASSESSMENT IN PERSPECTIVE

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5.1 Calculated radionuclide release rates and doses

5.1.1 Releases from the engineered barrier system

Table 5.1.1 shows, for different safety assessments, the six nuclides that are released at the highest maximum rates from the engineered barrier systems, as well as the rates themselves (in Bq a^{-1} per waste package). H12 is compared with H3 and Kristallin-I as examples of assessments of vitrified waste disposal. SITE 94 and TILA 99 are taken as examples of spent fuel disposal.

Tab. 5.1.1: The rates of release of the six nuclides released at the highest maximum rates from the EBS (in Bq a^{-1} per waste package) in different safety assessments

H12	H3[1]	Kristallin-I[2]	SITE 94	TILA 99[3]
Cs-135 (1×10^5)	Se-79 (2×10^7)	Cs-135 (7 x 10^4)	Cs-135 (2×10^5)	C-14 (2×10^6)
Nb-93m + Zr-93 (6×10^3 *)	Cs-135 (2×10^7)	Tc-99 (6 x 10^3)	C-14 (2×10^5)	Sn-126 (3×10^5)
Tc-99 (2×10^3)	Sn-126 (4×10^5)	Ra-226 (9 x 10^2)	I-129 (1×10^5)	Ra-226 (2×10^5)
Sn-126 (2×10^3)	Zr-93 (4×10^4)	Sn-126 (7 x 10^2)	Ra-226 (1×10^5)	Cl-36 (2×10^5)
Th-229 + U-233 (7×10^2 **)	U-233 (5×10^3)	Se-79 (2 x 10^2)	Cl-36 (7×10^4)	Ni-59 (1×10^5)
Th-230 + Pb-210 (4×10^1)	Th-230 (4×10^3)	Zr-93 (4 x 10^1)	Ni-59, Th-230 (2×10^4)	I-129 (5×10^4)

* : Nb-93m : 5.4×10^3 Bq a^{-1}, Zr-93 : 5.4×10^2 Bq a^{-1}
** : Th-229 : 6.8×10^2 Bq a^{-1}, U-233 : 2.6×10^1 Bq a^{-1}

Notes:
1 From Table 4.5-6 in H3
2 E. Curti (pers. comm. 2000) and Nagra (1994b)
3 From Table 11-21 in TILA 99 - overpack "disappears" at 10^4 years, non-saline groundwater.

Some notable differences are:

- The greater prominence of Se-79 in the EBS release in H3 compared to H12. Among the reasons for this is that the effect of shared elementary solubilities is taken into account in H12, whereas it is conservatively neglected in H3. The inventory of stable selenium is an order of magnitude higher than that of Se-79, which has the effect of reducing the concentration of Se-79 in solution required in order to reach the selenium solubility limit. Furthermore, the solubility limit for selenium in H12 is more than two orders of magnitude lower than in H3.
- The generally lower release rates in H12 compared to H3. Among the reasons for this difference is that, in H12, release from the outer boundary of the buffer is limited by the rate of groundwater flow through the excavation disturbed zone. In H3, this flow rate is effectively set to infinity (a conservative, zero-concentration boundary condition for diffusion through the buffer is used).
- The prominence of nuclides with the potential to form gases in the results of the assessments of direct disposal of spent fuel (SITE 94 and TILA 99). In assessments that address vitrified high-level waste (H12, H3 and Kristallin-I), these nuclides have very small inventories, since they are largely removed during reprocessing.
- The prominence of Cs-135 in all the assessments, with the exception of TILA 99 in the case of non-saline groundwater (where, at 4.6×10^4 Bq a^{-1}, it would be seventh in the list of nuclides). In all assessments, caesium is assumed to have a solubility limit that is high enough not to be exceeded. The degree of sorption assumed in the buffer is higher in TILA 99 in the case of non-saline groundwater (0.2 m^3 kg^{-1}) compared to the other assessments (all 0.01 m^3 kg^{-1}), although, in the case of saline groundwater, it is comparable to the other assessments (0.04 m^3 kg^{-1})[37].

In order to examine the relative contributions of the various components of the engineered barrier system, performance is examined in terms of:

- The ratio of the maximum release rates of different nuclides from the EBS to the maximum release rates from the waste form
- (for H12 and Kristallin-I, since this information is available) The inventories of different nuclides in the various EBS components as a function of time.

The ratios of release rates for the nuclides giving the highest EBS releases are presented in Table 5.1.2. The table again shows the relative ineffectiveness of the buffer at limiting Cs-135 release, except in the case of TILA 99.

37 It should be noted that, in SITE 94, 10 % of the Cs-135 inventory is assigned to the "gap-release" fraction, which is released instantaneously from a breached overpack. The remaining 90% is assigned to the more slowly released "grain-boundary release" fraction. None, however, is released congruently with fuel matrix dissolution. In TILA 99, 6 % is assigned to the instant-release fraction, with the remainder released congruently with fuel matrix dissolution

Tab. 5.1.2: Comparison of the ratio of the maximum release rates of different nuclides from the EBS to the maximum release rates from the waste form for different safety assessments

Nuclide[1]	H12 [2]	H3 [3]	Kristallin-I [4]	TILA 99 [5]
Cs-135	0.42	0.62	0.21	1.5×10^{-5}
Nb-93m (Zr-93)	5.1×10^{-3} (4.8×10^{-4})	-	(2.3×10^{-5})	(3.8×10^{-7})
Tc-99	3.1×10^{-4}	-	4.8×10^{-4}	1.0×10^{-6}
Sn-126	3.3×10^{-3}	-	9.5×10^{-4}	1.3×10^{-4}
Th-229 (U-233)	2.2×10^{-3} (8.8×10^{-5})	-	2.7×10^{-4} ($2.9\ 10^{-4}$)	4.7×10^{-2} (4.4×10^{-3})

Notes:

1 The release rates of Nb-93m and Th-229 from the EBS are determined, at least in part, by ingrowth from their longer-lived parents. Ratios of release rates for the parent nuclides are shown in parentheses.

2 Release rates from the waste form were obtained by multiplying glass dissolution rate by the concentration of the nuclide in the waste form and glass surface area.

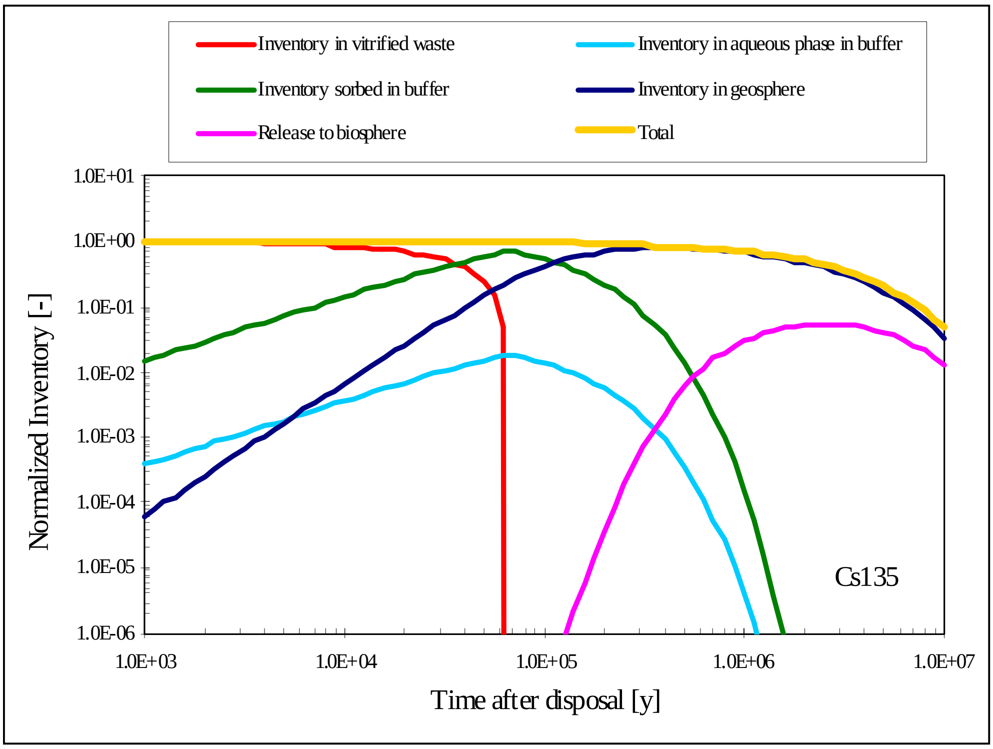
3 In H3, release rates from the waste form were not calculated explicitly except for Cs-135. Release rate of Cs-135 was obtained by multiplying the glass dissolution rate by the amount of the nuclide in the waste form and glass surface area. While, for other nuclides, instantaneous dissolution from the waste form and solubility-limiting concentration at the inner boundary of the buffer region were introduced.

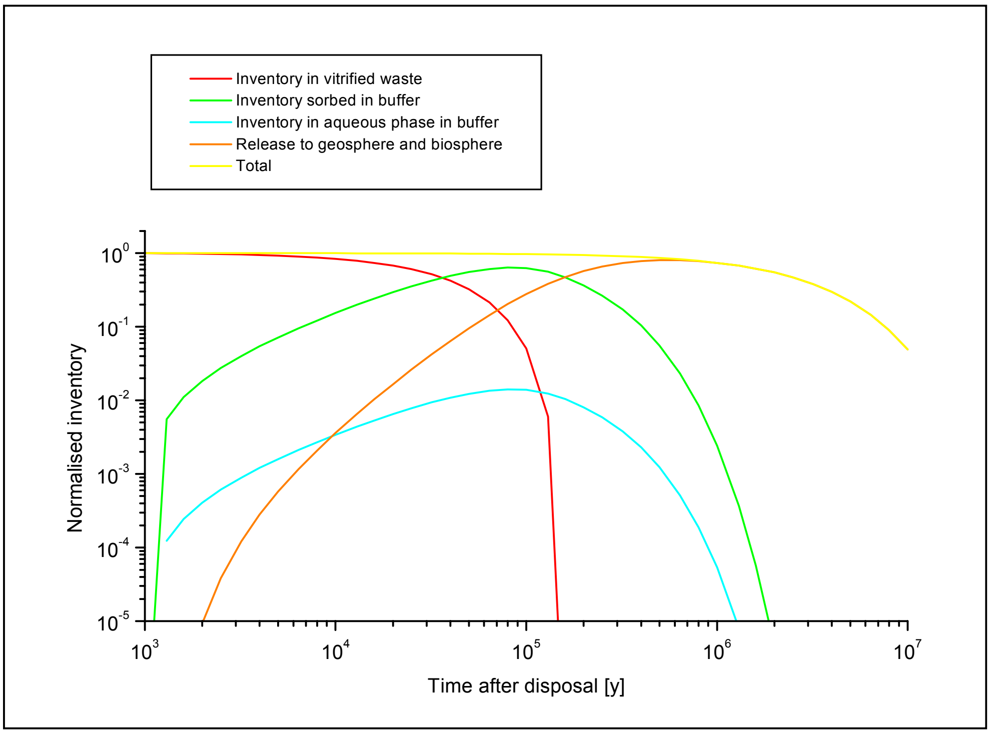
4 E. Curti (pers. comm., 2000)

5 From Table 11-21 in TILA 99 - overpack "disappears" at 10^4 years, non-saline groundwater.

Figure 5.1.1 shows the distribution of inventories of two example nuclides, Cs-135 and Np-237, between the vitrified waste form, precipitate (due to the solubility limit being exceeded), the buffer and external to the EBS (release to geosphere), as a function of time for the H12 and Kristallin-I assessments. In the case of H12, the inventory in the buffer is shown as aqueous and sorbed phases. In addition, in the case of H12, the inventory external to the EBS is partitioned between geosphere inventory and inventory released to the biosphere.

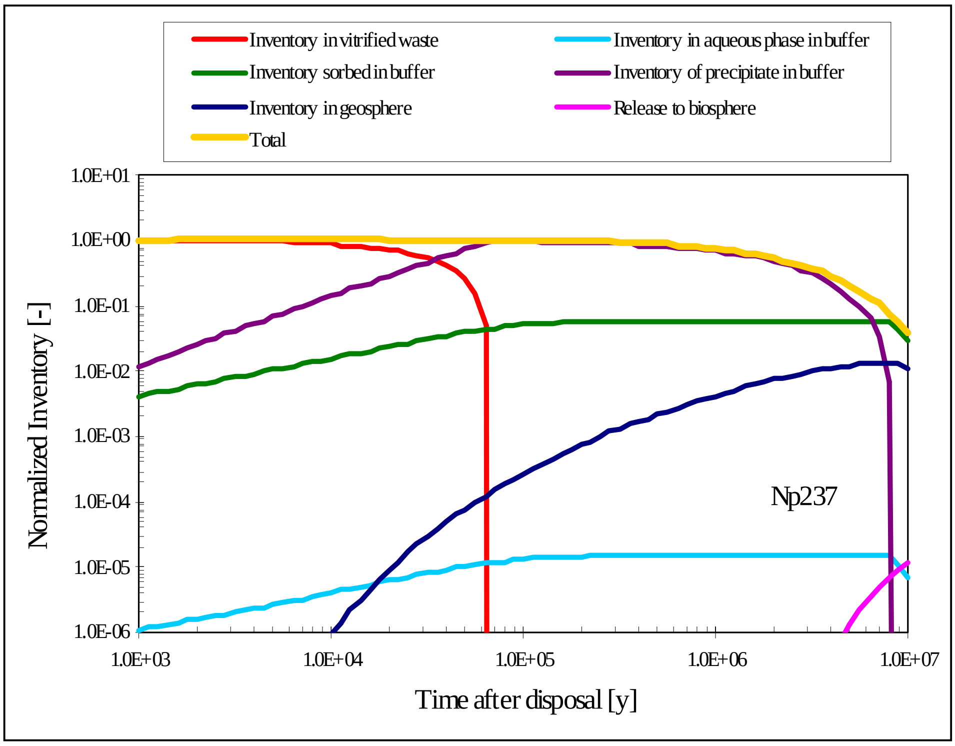
The figure shows the very similar behaviour of Cs-135 in the two assessments. In particular, it shows that a large part of the inventory is transferred from the EBS to the geosphere before substantial radioactive decay can occur. Np-237 shows qualitatively similar behaviour in the two assessments. In contrast to Cs-135, this nuclide is contained in the EBS (initially in the glass, and then later, predominantly as precipitate) sufficiently long for significant decay before transfer to the geosphere. The amount of decay that occurs before release to the geosphere is greater in the case of Kristallin-I, which also shows a lower inventory in the bentonite. This reflects the higher Np solubility limit assumed in H12 (2×10^{-8} mol dm^{-3}, compared to 10^{-10} mol dm^{-3} in Kristallin-I, see Table 4.5.6).

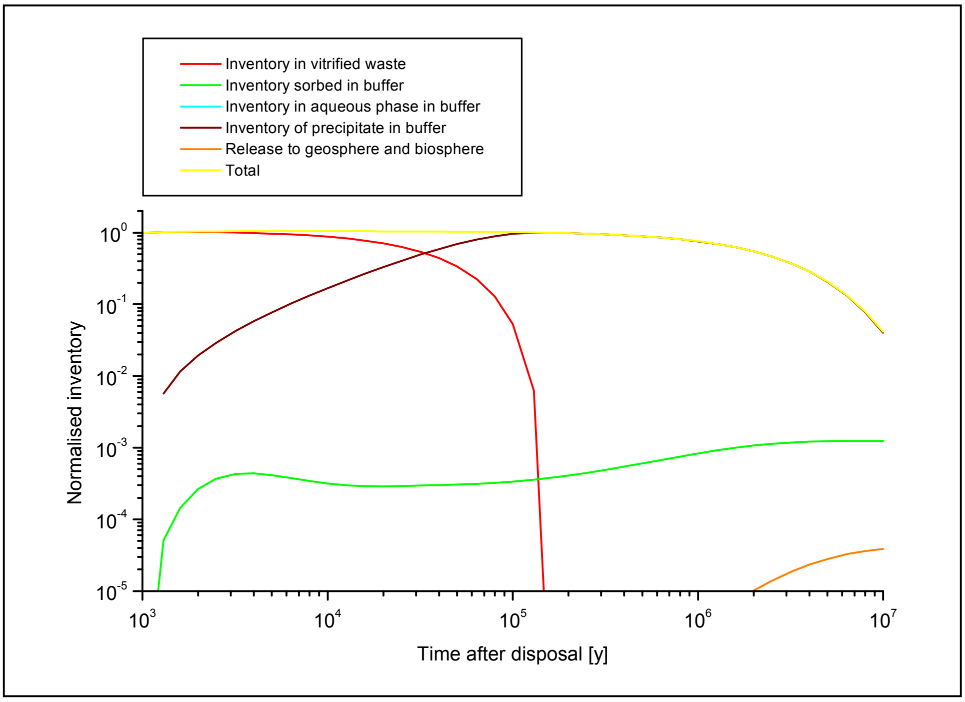




Note: Separate inventories of Cs-135 in the geosphere and released to the biosphere are not available for Kristallin-I (lower graph).

Fig. 5.1.1a: The distribution of inventories of Cs-135 as a function of time for the H12 (upper graph) and Kristallin-I (lower graph) assessments. The inventory is normalised to an initial value of unity





Note: Separate inventories of Np-237 in the geosphere and released to the biosphere are not available for Kristallin-I (lower graph).

Fig. 5.1.1b: The distribution of inventories of Np-237 as a function of time for the H12 (upper graph) and Kristallin-I (lower graph) assessments. The inventory is normalised to an initial value of unity.

5.1.2 Geosphere releases

In order to examine the performance of the geosphere, and its contribution to the attenuation of nuclide releases, relative to that of the EBS, the following are considered:

- The ratio of the maximum release rates of different nuclides from the geosphere to the maximum release rates from the EBS
- (for H12 and Kristallin-I, since this information is available) The decay of inventories of different nuclides in the geosphere compared to that in the EBS.

The ratios of release rates for the nuclides giving the highest EBS releases in H12 are presented in Table 5.1.3.

The table shows the very high effectiveness of the H3 geosphere compared to the other assessments and, by contrast, the ineffectiveness of the TILA 99 geosphere in attenuating the release maxima.

Referring to sections 4.6.1 and 4.6.2, differences in properties related to groundwater flow (in particular, the "transport resistance") and matrix diffusion do not appear sufficient to explain the very marked differences in geosphere effectiveness. More significant are the differences in the sorption properties of the matrix. For example, in H3, although broad ranges of *Kd* are given for different elements, the upper bound of the range is used in geosphere transport calculations (section 4.5.6.2 in PNC, 1992). H12 and the other assessments in Table 5.1.3 are more conservative in their selection of *Kd*.

The relative effectiveness of the EBS and the geosphere can be illustrated by calculating the degree to which the inventory of each nuclide decays within a particular barrier or barriers, as shown in Table 5.1.4 in the cases of H12 and Kristallin-I.

The table illustrates the high degree of effectiveness of the overall multi-barrier system, as evaluated in both assessments. With the exception of Cs-135, less than 5 % of any nuclide passes from the EBS to the geosphere, and over 99 % decays within the combined EBS and geosphere.

Cs-135, with its relatively long half life and low sorption, decays relatively little in the multi-barrier system, before its release to the surface environment.

Tab. 5.1.3: Comparison of the ratio of the maximum release rates of different nuclides from the geosphere to the maximum release rates from the EBS for different safety assessments

Nuclide	H12		H3 [1]	Kristallin-I	TILA 99
	host rock	host rock + MWCF			
Cs-135	1.2×10^{-2}	9.1×10^{-3}	8×10^{-12}	0.66	0.34
Nb-93m (Zr-93)	3.3×10^{-3} (3.3×10^{-2})	2.3×10^{-3} (2.3×10^{-2})	- [2] (1.1×10^{-19})	0.18	(-)
Tc-99	7.3×10^{-4}	3.1×10^{-8}	3.3×10^{-6}	8.8×10^{-3}	0.21
Sn-126	7.2×10^{-5}	2.8×10^{-11}	$< 2.4 \times 10^{-21}$	1.7×10^{-4}	0.68
Th-229 (U-233)	2.4×10^{-3} (6.1×10^{-2})	3.0×10^{-4} (7.6×10^{-3})	- [2] (4.0×10^{-8})	1.3×10^{-2} (9.7×10^{-3})	0.040 (0.86)

Notes:
1 From Table 4.5-6 in H3
2 In H3, Nb-93m and Th-229 were not selected as nuclides for consideration in nuclide migration calculations. For Th-229, the dose estimation has been carried out using the release rate of the parent nuclide from geosphere.

Tab. 5.1.4: The percentage of selected nuclides decaying within the EBS, and within the EBS and geosphere combined (data taken from Table 6.2.4 of Nagra (1994b) and Table 6.4-1 of JNC (2000d))

Nuclide	Decay within the barrier system (%)			
	EBS		EBS + Geosphere	
	H12	Kristallin-I	H12 (host rock)	Kristallin-I
Cs-135	10.98	5.48	84.90	12.33
Nb-93m (Zr-93)	100.00 (86.89)	(99.60)	100.00 (99.38)	(100.00)
Tc-99	95.03	98.60	100.00	99.93
Sn-126	92.57	99.56	100.00	100.00
Th-229 (U-233)	99.69 (99.45)	100.00 (99.97)	100.00 (100.00)	100.00 (100.00)

5.1.3 Indicators of overall performance

Individual dose is used in H12 as the primary indicator of overall system performance. As in many other counties, the use of dose as a performance indicator is supported by regulatory guidelines, in the Japanese case AEC Guidelines (AEC, 1997). The guidelines state, however, that, although the " ... *overall safety of the geological disposal system should be evaluated in terms of radiation dose as a primary indicator* ...", other indicators "... *should be used in order to compare the consequences* *of geological disposal with natural radiation levels thereby minimising uncertainties associated with future human activities*".

Figure 5.1.2 shows the calculated annual individual dose as a function of time for the Reference Cases of H12 and eight other HLW and SF assessments. Doses are compared with the range of natural radiation exposure in Japan (approx. 900 to 1200 μSv a^{-1}) and to the range of regulatory guidelines in various countries, excluding Japan (100 to 300 μSv a^{-1}). The nuclides that contribute most to dose at different times are also indicated.

Whereas previous subsections have generally considered releases from individual waste packages, the results in Figure 5.1.2 are based on the entire repository inventories in the cases of H12, H3, Kristallin-I, Projekt Gewähr and AECL EIS[38]. SITE 94 considers a single defect in an overpack, which is present at the time of disposal. In the case of TILA 99 and TVO 92, results are shown for the case of a single "disappearing" overpack at 10^4 years following disposal. The case for SKB 91 is less simple since the assessment is in part probabilistic including stochastic overpack failure. The results presented represent the sum of releases from about 6 waste packages that fail after 1000 years, where each plume of radionuclides migrates independently through the geosphere (see Neall (1994), section 4.5.2, for a full description of the case presented).

All the SA results shown in Figure 5.1.2 have dose maxima in the range 0.01 - 10 μSv a^{-1}, except for the older Projekt Gewähr, which has a significantly lower maximum at 10^{-5} μSv a^{-1}. For the five spent fuel assessments, SKB 91, SITE 94, TVO 92, TILA 99 and AECL EIS, the dominance of I-129 is clear. As mentioned previously, this is due to the presence of this highly mobile nuclide in the instant release fraction of the spent fuel. There is no comparable phenomenon in the case of vitrified HLW, so these assessments are dominated by long-lived nuclides which are not limited by solubility and/or sorption such as Cs-135 and Se-79. At very long times, > 1 Ma, actinide nuclides such as Pa-231 come to dominate, as the flux of these from the near field is supported by the dissolution of relatively insoluble parent nuclides (e.g. U-235 in the case of Pa-231) after the waste glass is completely dissolved.

38 In the case of AECL EIS, although all the waste packages fail before 10 000 years, the results are dominated by releases from sections of the repository closest to the fault which acts as a conduit to the biosphere. Releases from the other repository sections do not reach the biosphere within the assessment period

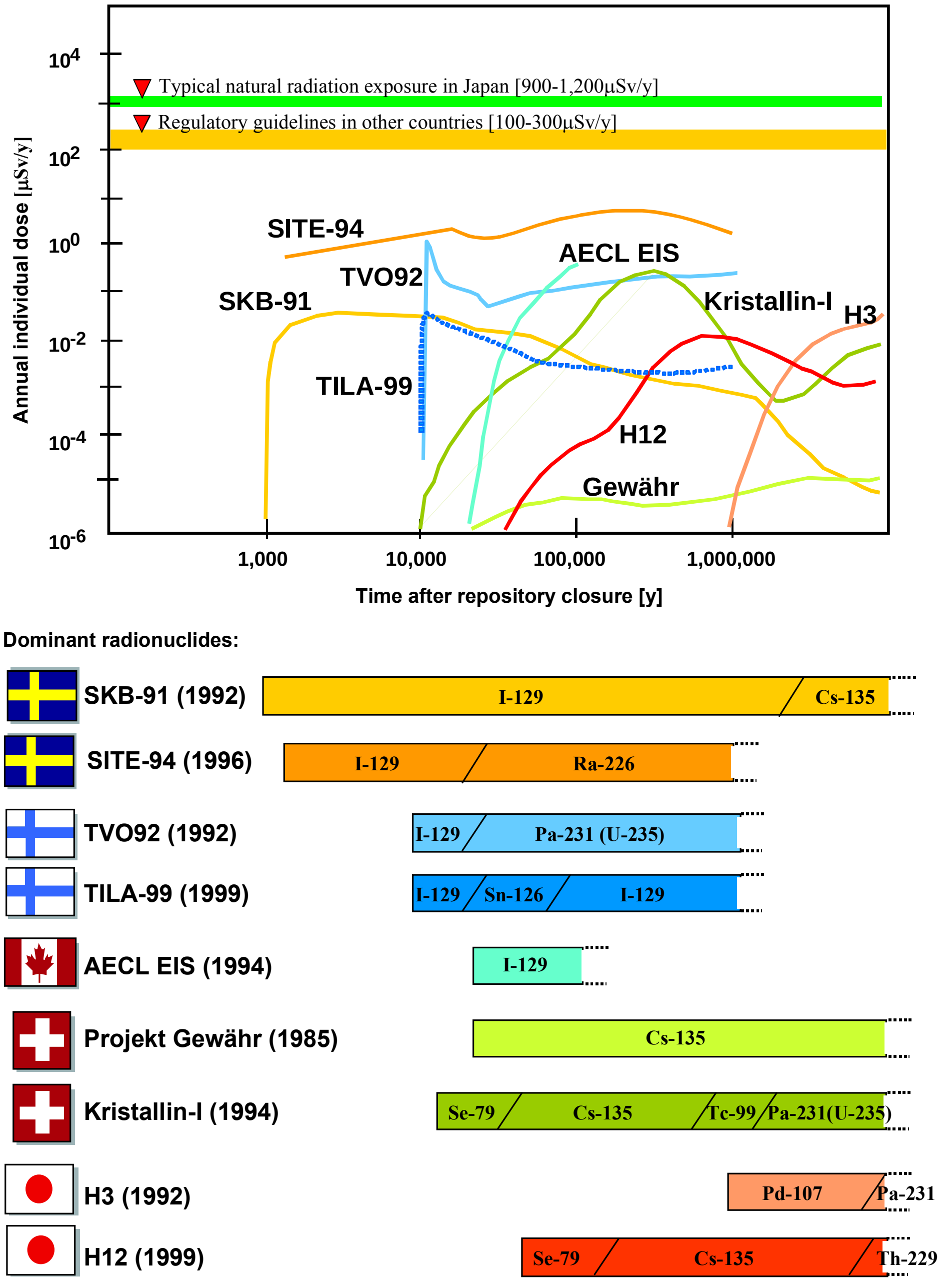


Fig. 5.1.2: Calculated results from a range of safety assessments

It is interesting to note the significantly higher maximum for Kristallin-I compared to its predecessor, Projekt Gewähr. As discussed in Neall (1994), this is largely due to the more realistic geosphere models used in the later assessment. For the Japanese assessments, the maximum doses arise much earlier in H12 than in H3. Again this is partly due to the modified description of the geosphere but also due to significant changes in the element-specific data for Cs and Se between the assessments.

Alternative performance indicators used in H12 include:

- Concentration and flux of specific elements or nuclides
- Radiotoxicity index.

These may be more indicative of the isolation capability of a disposal system and the potential risks of radioactive waste than individual dose alone, over the very long time scales of safety assessment (IAEA, 1994).

Alternative indicators studied in other national programmes include, in Switzerland, a comparison of the calculated concentrations of nuclides released from a repository with those of natural radionuclides occurring in groundwater (Neall, 1994). In Sweden, natural radionuclide fluxes resulting from groundwater, denudation by glaciers, and weathering have been calculated and the results compared with Swedish safety limits (Miller et al., 1996).

5.2 Key factors that provide safety

H12, Kristallin-I, SITE 94 and TILA 99 are examples of safety assessments that use a combination of reasoned arguments and sensitivity analyses to identify the key factors of the disposal concepts that provide safety. Table 5.2.1 presents a comparison of key factors identified either implicitly or explicitly in these different assessments.

Dilution in surface and near-surface water is also a key factor affecting the calculated doses, although this is not defined as a safety factor in the assessments.

All of the assessments identify low solubilities in the near field, and a fine pore-structured buffer ensuring colloid filtration and diffusion-dominated transport of solutes, as key safety-relevant factors. Low groundwater flow, physical stability and favourable and stable geochemical conditions in the near field are also recognised as important in ensuring the longevity of the engineered barriers.

The assessments differ in the importance attached to the time during which the overpacks prevent water ingress, and radionuclide egress, and to the nuclide release rate from the waste form. In assessments that consider vitrified waste, these factors are found to be of secondary importance, whereas they are of key importance in assessments that consider the direct disposal of spent fuel. There are two main reasons for this difference:

- In assessments that consider the direct disposal of spent fuel, the maximum calculated doses are dominated by I-129, a proportion of which is assumed to be released instantaneously from the fuel. By comparison, in assessments that consider the disposal of vitrified waste, I-129 is effectively absent[39] and all nuclides are considered to be released congruently with waste form dissolution

39 It is either released to the environment during reprocessing or (nowadays) immobilised in long-lived ILW

Tab. 5.2.1: Key factors that provide safety identified in different safety assessments

Factors	H12, Kristallin-I	SITE 94, TILA 99
Lifetime of unbreached overpack	Secondary importance	Highly important (although initial defect or "disappearing overpack" assumed in calculational cases)
Nuclide release rate from waste form/spent fuel	Secondary importance	Highly important – especially "instant release fraction" of iodine
Favourable chemical conditions in the buffer (near field solubilities and sorption)	Low solubilities of some nuclides highly important; high sorption – secondary importance	Highly important
Fine pore structure of bentonite (colloid filtration, diffusion-dominated transport of solutes)	Highly important	Highly important
Low groundwater flow around the near field	Protection of bentonite from physical erosion and chemical alteration highly important; limitation of radionuclide release identified as highly important in H12, but is of secondary importance in Kristallin-I.	Protection of bentonite from physical erosion and chemical alteration, and limitation of radionuclide release highly important
Physical stability and favourable and stable geochemical conditions in near-field rock	Important for longevity of engineered barrier system	Important for longevity of engineered barrier system
Low transmissivity transport pathways in geosphere	Highly important for operation of geosphere transport barrier	Highly important for operation of geosphere transport barrier
Conditions favourable to geosphere retardation processes (matrix diffusion and sorption)	Highly important for operation of geosphere transport barrier	Highly important for operation of geosphere transport barrier

- In assessments that consider the direct disposal of spent fuel, the dissolution rate of the fuel is usually influenced by alpha radiolysis at the fuel surface – the longer water ingress is prevented, the lower the radiation field at the fuel surface and the slower the dissolution rate of the fuel, whereas, in assessments that consider the disposal of vitrified waste, the dissolution rate of the waste form is considered to be independent of overpack lifetime.

The assessments all identify low groundwater flow rates (and, in particular, low transmissivity pathways in the geosphere) and conditions favourable to geosphere retardation processes as factors that are important to the effectiveness of the geosphere transport barrier. H12, Kristallin-I and SITE 94 recognise that further site-specific information regarding these factors will be required in order to take full advantage of this barrier.

5.3 Support for the long-term operation of key factors

5.3.1 Support from observations of natural systems

Qualitative evidence that the key factors that provide safety will operate as expected is available from observations of natural systems that provide analogies to the disposal system of interest. Natural (and anthropogenic) analogues complement long-term laboratory and field experiments (section 5.3.2), and have the advantage that they provide information on the operation of processes over timescales similar to (or often exceeding) those of relevance to safety assessment.

Despite the existence of a large body of work (see, for example, the reviews of Pearcy and Murphy, 1991; Miller et al., 1994, 2000), it has frequently been stated (e.g. Smellie and von Maravic, 1994; Smellie et al., 1997) that the use of natural analogue (NA) data in SAs has been minimal. Until recently, few SAs had specifically referred to the direct use of NAs with the exception of Kristallin-I where NA input appears in sections relating to:

a) Confidence building

b) Development of assessment methodology and models

c) Engineered barrier longevity

d) Matrix diffusion depths and

e) Groundwater stability

(see also Alexander and McKinley, 1999, for details). Indeed, in some other SAs, no reference was made of NAs at all (e.g. Yucca Mountain: Wilson et al. 1994). In other cases, such as SKB 91 and H3, some mention is made of the use of natural analogues in supporting concepts and models.

More recently, as in TILA 99, numerous references have been made to NA data which are consistent with the data used in the SA calculations for processes such as matrix diffusion, far-field retardation and spent fuel dissolution. Curiously, although the SR 97 Processes Report (SKB, 1999b) is specifically structured to include NA input (but not, surprisingly, field data - see below) to each process discussed, most entries read simply "not applicable". Only in a few cases, such as for matrix diffusion or far-field retardation, is NA input discussed at all and even here it tends to be very generalised with no clear indication of quantitative input. Considering the large body of NA work produced by SKB which is clearly of direct relevance to SA (e.g. Hallberg et al. 1987; Cramer and Smellie 1994; Bruno et al. 2001), it is assumed that the non-application of NA data in SR 97 is simply a reflection of the views of the report authors.

In comparison with the above reports, the use of NA studies is widespread in H12, supporting conceptual models and providing bounding limits on various processes and mechanisms. Specific examples from studies in Japan and worldwide are used to qualitatively support the fundamental ability of the geological environment to retain (or trap) radionuclides with reference being made to the Cigar Lake, Oklo, Koongarra, Osamu Utsumi and Tono uranium ore bodies. Of particular interest here is the use of the Tono study site to show that, despite experiencing uplift, erosion and faulting (all mechanisms of some significance to the siting of a repository in Japan), the Tono uranium ore body has been relatively stable for over 10 Ma, leading to the statement that "This indicates that the conclusions of the SA analyses carried out for a Japanese repository are at least credible." (H12 Supporting Report 3).

Other examples include:

- Comparison of corrosion rates of archaeological materials worldwide and those used in the SA calculations, showing that the SA data are conservative
- Qualitative comparison of the thermal histories of natural bentonites with that expected in the repository, indicating that thermal alteration will not be an issue for the bentonite backfill.

Quantitative use is made of NA data on matrix diffusion (particularly from JNC's Kamaishi Test Site in Japan, see Yoshida et al. 2000, for details) and information on natural fluxes of radionuclides is used to place the potential releases from the repository in context.

5.3.2 Support from long-term experiments and field studies

As mentioned previously (section 2.5), H12 has been extensively supported by an in-house experimental R&D programme, which started during H3, partly as a response to questions raised by that assessment. Examples of notable experiments are BENTFLOW, which was used to assess bentonite extrusion and erosion in fractures (Kanno et al. 1999; Matsumoto et al. 1997), and the EBS seismic stability tests in which a scale-model of the EBS, including overpack and bentonite buffer, was subjected to a simulation of an earthquake in order to test the response of the different parts of the system. The aim here was to support the claim that seismic disturbances will not cause the degradation of the EBS, for example by movement of the dense overpack within the compacted bentonite leading to thinning of the bentonite or to voids within the bentonite. In both experiments, the results were used to justify excluding FEPs from quantitative analysis.

However, long-term laboratory or field experiments to support the SA are less numerous and references to the results, either in terms of data taken over to SA models or confirmation of conceptual models or confidence building, harder to find in most assessments.

One example is the long-term glass leaching experimental programme carried out as a Japanese – Swedish – Swiss collaboration (SKB/JSS, 1987; Björner, 1988) to justify the use in SA of a low, long-term leach rate for the glass matrix, by demonstrating the decline in leach rate from the high initial value previously measured for short-term experiments. Data from these and other experiments are interpreted in Ohe et al. (1991) to define a justifiable long-term leachrate, so that their use is not explicit in H12 or, indeed, in Kristallin-I.

Another area which has received much attention is solute transport in the geosphere. Experimental programmes have been carried out at the Äspö URL in Sweden and the Grimsel Test Site (GTS) in Switzerland – as international collaborations in both cases.

With respect to the GTS migration experiment (MI), its most important use has been the development of testing methodologies and the application of those methods to confidence building within SA. Kristallin-I specifically mentions the contribution of the MI experiment to model testing in general. Further, it was noted that "...the results provide confidence in the dual-porosity concept[40] as an appropriate foundation for a

40 The dual-porosity concept is the description of fractured rock in which solute transport in the fractures maybe by advection but the matrix rock adjacent to the fractures is accessible only by diffusive transport. Effectively, the fractures are represented by well-connected porosity which can support flow, whereas the matrix is less well

model of transport in fractured porous media". In addition, it was noted that "...the model provides a satisfactory interpretation of the measured data and no evidence has been found which would indicate that processes relevant to safety assessment and not accounted for in the model are operating".

Some effort has gone into extrapolating data on retardation mechanisms from MI to repository relevant host rocks, but this has been limited in some instances. Although matrix diffusion has been identified as important and its effectiveness better assessed than previously by experiments, the diffusion constants for the rock matrix used in Kristallin-I were "...selected on the basis of a survey of (*laboratory*) experimentally determined diffusion constants for crystalline rocks. " (authors` italics). Also, no reference was made in Kristallin-I to evidence from the MI experiment when depths of accessible wall rock are considered, other than in the case of one parameter variation where data from MI supporting experiments are used to define a minimum depth of diffusion.

The work on investigating the connection between laboratory measured sorption data and field retardation has shown that, with enough background information on the flow field, it is possible to show reasonable agreement within the MI experiment between field and laboratory data. However, this has not yet influenced SA databases or been used in H12 or Kristallin-I.

5.4 Support for total system performance

Support for the concept of deep geological disposal as a means of isolating waste (at least for sparingly soluble elements, such as uranium, thorium and plutonium) from the surface environment over a very long timescale is provided by observations of uranium ore deposits, for example at:

- Cigar Lake, Canada (Cramer and Smellie, 1994)
- Krunkelbach, Germany (Hofman, 1999)
- Tono, Japan (Seo and Yoshida, 1994).

In these examples, the several hundred metre thick rock layer above the deposit has provided an effective barrier to transport, even though the some parts of the ore bodies may have come into contact with oxidising groundwaters and, in this respect, the situation is less favourable to isolation than would be expected in a repository.

The large uranium ore deposit at Cigar Lake is interesting because it is unusually rich (up to 55% uranium) and, despite its relatively shallow depth of 430 m, there is no detectable anomaly at the surface. The deposit lies at the junction of crystalline basement and overlying sandstones, and the uranium ore is surrounded by 5 – 30 m of clay, which may be considered analogous to the buffer.

The Tono uranium deposit is much shallower, approximately 150m below the surface, lying at the boundary between a Cretaceous granite and the overlying Mizunami sediments but despite faulting, which displaced the ore body, and uplift and erosion since 0.7Ma, there is little evidence for remobilisation of the uranium. In this case, it seems that the geochemical conditions around the orebody remain very favourable due to very slow groundwater access (Mizutani et al., 1992; Seo and Yoshida, 1994).

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connected and the pore waters are stagnant. The depth away from the fracture wall to which the matrix porosity is considered connected determines the matrix diffusion depth

Further examples of ore deposits that have been studied in the context of geological disposal are at Poços de Caldas, Brazil (Chapman et al., 1990, Nagra 1993) and Alligator Rivers, Australia (Duerden et al., 1992).

Support for the concept of isolating a range of elements, including fission products, is provided by the Oklo natural reactors in Gabon, where, about 1800 million years ago, nuclear fission reactions took place in fractures in Precambrian argillaceous sandstones and shales, where high-grade uranium oxide had been precipitated. The reactor core temperatures were in the order of several hundred degrees, and thus significantly higher than those expected in a repository. The data from Oklo suggest that there was very little radionuclide migration during the 800,000 years of reactor operation, despite the occurrence of hydrothermal circulation (Brookins, 1990), and during the 1800 million years thereafter.

Exceptions are the gases (e.g. iodine) and alkali elements (e.g. caesium), which have largely been lost from the uranium oxide, but retained, to some extent, in the neighbouring ferromagnesian clay minerals, which may be considered analogous to the buffer minerals. This is in agreement with the safety assessment modelling results discussed in above, which show that the EBS and host rock may provide less effective barriers for these elements than for other safety-relevant components of the inventory.

6 TIMESCALES

F.B. Neall

This section discusses two aspects of the timescales used in safety assessment:

- The reasons for the very long periods considered in many analyses
- The meaning of results from the far future, including a comparison of SA timescales with geological and historical timescales.

6.1 Long timescales

Long timescales for safety assessment are necessary for two main reasons:

- The potential hazard associated with HLW is very long-lived as many safety relevant radionuclides have very long half lives, in excess of 10^5 years. Figure 6.1.1 shows the decrease in radioactivity of the HLW over time after emplacement, compared to the radioactivity of the original quantity of uranium ore.
- The slow processes involved in release of radionuclides from the repository mean that a considerable amount of time is required for significant releases to occur. For example, since overpacks in H12 are designed to remain unbreached for at least 1000 years, and radionuclides must then travel from the HLW to the biosphere in order to cause exposure of a population, an assessment time much longer than 1000 years is required, especially if dose is used as a performance indicator.

It is usually also considered desirable to determine the maximum dose arising and the time at which this occurs[41]. It is also considered unethical to potentially expose future generations to doses which would not be currently acceptable, on the basis that the doses arose after some arbitrary time cut-off and thus were not calculated. The relevant Swiss regulatory guideline R-21 (HSK and KSA, 1993) states that "The release of radionuclides from a sealed repository ….shall at no time give rise to individual doses which exceed 0.1 mSv per year". This may lead to analyses being carried out over periods of the order of 10^6 years or longer, even if results from longer times are interpreted only qualitatively. However, the results of some analyses are presented with the calculated doses still rising at the cut-off (see Fig. 5.1.2). At the same time, it is necessary to realise that calculations can be carried out over timescales which are too great – the age of the earth is only around 5 x 10^9 years and calculations over about 10^7 years become meaningless due to the possibility of unpredictable large scale changes (see section 6.2).

Thus, the consideration of such long timescales requires explanation of the meaning of the results at very long times as, clearly, we can have no certain knowledge of human activities, climate and even the geological environment so far into the future.

41 Note that this is not currently the case in the USA, although some groups (e.g. National Academy of Science) are pushing for it

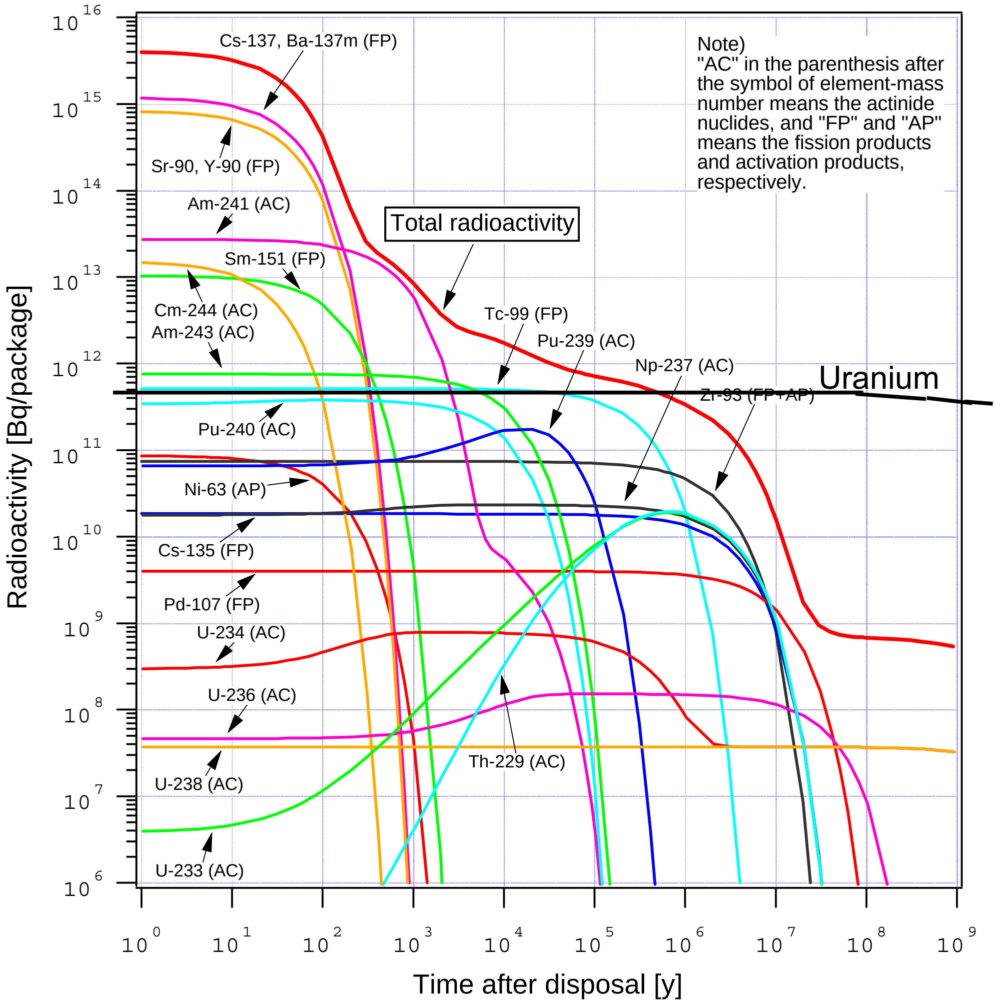

Fig. 6.1.1: Example of the changes with time in the radioactivity of one package of model vitrified HLW. The horizontal line marked "uranium ore" is the approximate radioactivity of the original uranium ore that was processed to provide the reactor fuel and which resulted in one package of vitrified HLW. The half lives of the U-235 and U-238 in the ore are sufficiently long that almost no decrease in radioactivity is apparent on this timescale

6.2 Meaning of results for the far future

The results of SA calculations should be considered as ".. *illustrations of potential future conditions and associated repository response..*" (see Chapman et al., 1995), rather than predictions. Scenarios which usually form the basis for calculations are a set of assumptions about the processes that will control radionuclide release from the repository. These assumptions can be changed to reflect different possible future evolutions of the geosphere, biosphere or the repository itself. Thus the impact of changes which occur over long periods can be investigated even if it is not possible to "predict" when or if they will occur.

Furthermore, when dose to an individual is used to indicate the performance of a repository system, it is a convenient way of turning calculated chemical releases into a radiological effect on a human population, using knowledge of current human behaviour. Doses, especially in the far future, are not predictions but indicators, since human behaviour is assumed to remain unchanged with time.

Hence, a dose versus time plot, as commonly used (e.g. Figs 2.3.7 and 5.1.2) to illustrate SA results, is showing a possible outcome for a particular set of assumptions about the processes involved.

The descriptions of the processes included (or processes excluded) are based on current information and understanding. For example, the detailed record of volcanism in Japan, and understanding of the factors which control it, allow prediction of the future position of volcanism over a period of at least 10^5 to 10^6 years or more (see H12 Supporting Report 1). Thus it is not necessary to take account of volcanism on the repository over the first 10^5 – 10^6 years, if a suitable site is chosen. On timescales greater than this, less predictable changes to the larger plate tectonic environment of Japan may take place which make it more difficult to extrapolate from the recent past. This means that, in theory, it is possible that the repository site could be affected by volcanism and less confidence should be placed on the calculated results at longer times. In practice, repository sites can be chosen which are very unlikely to be affected by, for example, volcanism, on even very long timescales since changes to plate tectonics are not random nor sudden (on a historical timescale) but slow developments from existing patterns. Thus it is possible to conjecture a range of future possibilities and use this to define areas least likely to be affected.

Assumptions about future uplift or subsidence rates, large-scale groundwater flow regimes and faulting are also based on extrapolation of the patterns of recent (in geological terms) geological evolution, justified by our understanding of fundamental mechanisms such as plate tectonics.

If geological evolution gives rise to uncertainty about the applicability of results at long times, > 10^5 years, climatic changes can operate on significantly shorter timescales. For example, the next ice age is predicted to occur within about 10,000 years. Although this is unlikely to affect most of Japan directly (mainly the mountainous areas in Honshu and Hokkaido), changes to global oceanic or atmospheric currents could result in local effects and the lowering of sea level could result in increased surface erosion. These changes could affect the type of human activities, local groundwater movement (if rainfall were affected, for instance) and surface processes, and thus the appropriateness of the biosphere model used to assess the radiological impact.

Climatic changes due to man-made global warming potentially also introduce relatively unpredictable changes on a shorter timescale. Thus the timescales over which the biosphere conditions can be bounded with any certainty are much shorter than for geological conditions, less than 10,000 years. But this itself is very long compared to timescales for predicting human activities: the increasing rate of technological, and societal, change means that human activities 100 years in the future are as unpredictable to us as the present day would have been to Meiji-era Japan or pre-Great War Europe.

Thus, even before the overpacks have failed in the closed repository, the assumptions used to calculate the radiological exposure of a human population to the eventual radionuclide release may be no longer appropriate. However, use of biosphere models still allows conversion of radionuclide releases to doses in a way which can take account of human behaviour. This is considered preferable to presentation of results only as radioactivity releases or chemical concentrations which, due to the different radionuclides and the way they interact differently with the body, are hard to compare in terms of significance to a human population.

For these reasons, other ways of illustrating repository performance, especially at long times, have been increasingly investigated. These would include use of radiotoxicity index instead of dose, determination of the distribution of radionuclides in the multi-barrier system components over time, comparison with naturally occurring radionuclide concentrations etc. (see, for example, Neall, 1994 or H12 Supporting Report 3)

Figure 6.2.1 shows a comparison of "events" in repository evolution as calculated in a safety assessment such as H12 and events which have taken place in human and geological history, illustrating the large time span necessarily involved in safety assessment analyses.

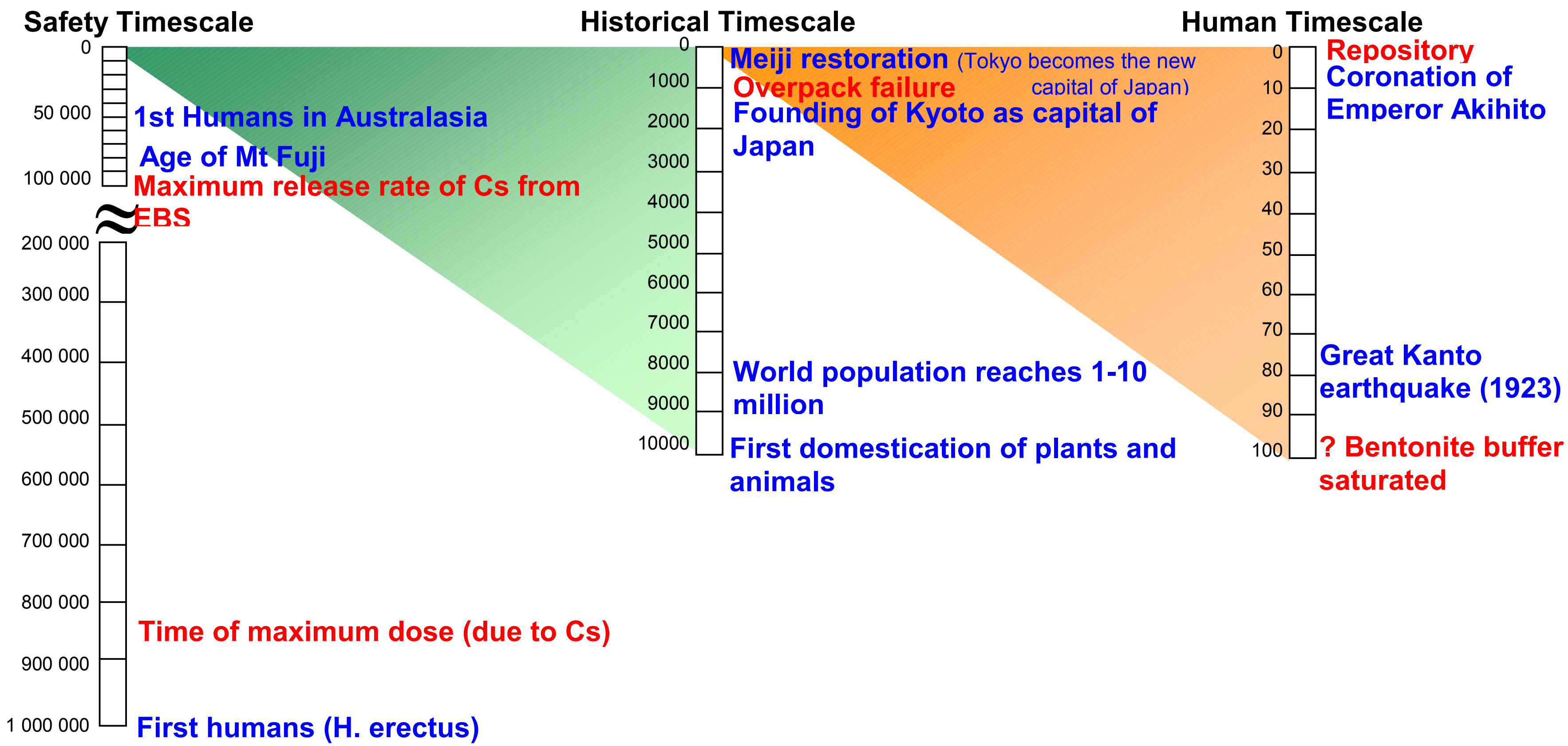

Fig. 6.2.1: **Comparison of timescales - SA timescales of greater than 1 million years are common but hard to grasp compared to the timescale of a few human generations that is more usual in normal life. SA timescales are also often plotted on a logarithmic scale which adds to the difficulty of comparison. This figure shows events on a linear timescale to try to illustrate how human (100 years), historical (10,000 years) and geological/SA (1 000,000 years) timescales can be linked. It should be noted that the timescale for events of relevance to a HLW repository (red text) is reversed (i.e. events will occur in the future) compared to that for actual historical events (blue text).**

7 RESULTS OF H12 FROM THE WIDER PERSPECTIVE

R.A. Klos, F.B. Neall

7.1 Public perception of risks associated with radioactive waste disposal

In the decades following the discovery of radioactivity, public perceptions of this phenomenon were generally extremely favourable. Highly radioactive sources were handled very casually and medical implications were seen as positive (e.g. "radium" ointment, radon inhalatoriums). Even the first development of nuclear weapons was perceived simply as providing more powerful explosives and widespread civil engineering use of such explosives was foreseen taking place in parallel to the development of nuclear power as a clean and cheap energy source.

A major change in public perception can probably by traced back to the mid/late twentieth century when a number of accidents at nuclear facilities dented public confidence in the nuclear industry (e.g. Windscale, 1957; Three Mile Island, 1979; Chernobyl, 1986). In Japan, confidence in the nuclear industry has been further reduced following a series of accidents at nuclear processing sites (e.g. Tokai, 1999). These accidents in Japan must also be viewed in the context of the atomic bombing of Hiroshima and Nagasaki in 1945. In addition, the growing awareness of the long-term health risks of low levels of radioactivity (carcinogenesis, tetragenesis) coincided with the expansion of the "environment movement". These experiences generate suspicions which apply to just about all matters nuclear throughout the world, something not helped by the manner in which scientists attempt to communicate safety.

Radiation routinely scores the highest in terms of perceived hazard (e.g., Wynne, 1989; Wiedemann, 1997) when compared to other potential risks, such as from toxic chemicals, infectious diseases etc., and this may be attributed to a combination of "dread" and lack of "familiarity". The main factors giving rise to concern about radiation hazards are that radioactivity has the potential to cause harm over long timescales and that its effect is unseen and not easily detectable. Concerns are compounded by radiation exposure being almost totally outside the control of the individual.

In addition, most of the general public's encounters with the concepts of radiation, nuclear power, radioactive waste and its disposal arise via the media. Mays and Poumadere (1996) note that even when reporting a serious and wholly constructive political dialogue in the search for a site for waste disposal research facility in France in the early 1990s, media reports invariably employed archive material from an earlier and abortive phase of the programme which led to civil conflict and "grave disorder". Members of the media were apparently conditioned by an expectation of public hostility and perhaps responded to the need for "a good story"

The debate in Japan is complicated by there being no precise word for risk[42]. Douglas (1992) speculates that the effect of this is to promote discussion in terms of moral and political concerns directly. To some extent this is mirrored elsewhere since to attach a "risk" to something is, in the mind of the public, to label it as inherently dangerous and

42 Risk is usually translated in Japanese directly as risuku or kikensei and is applied to the concept but the meaning is broad. In contrast, words for danger (kyofu or kiken) and damage (songai) have very precise meanings. "Probability" is also well defined (as in "probability analysis" – kakuritsuronteki-kaiseki) but this meaning is more in technical than common usage

often the word "risk" is used as a means of challenging faceless and remote authority. As such, "risk" is always employed as a political concept and not as a precisely defined mathematical quantity in the form employed in safety assessments and as set out in regulations designed to safeguard public health.

As noted above, the public associate "nuclear" and "radioactivity" with very negative concepts and because of this, it is likely that concern about radioactive waste disposal is first perceived in terms of "dread" and "unfamiliarity" derived from these concepts. Greater public understanding of the issues and, in particular, wider appreciation of the background to waste disposal to clarify the need for disposal, could lead to increased familiarity and public confidence. There is also a requirement to demonstrate that the risks associated with radioactive waste disposal arise from current power generation and medical uses of radiation and are not a wholly new form of risk.

Part of the problem undoubtedly reflects the unusual nature of radioactive waste disposal; in most major engineering projects such as bridge construction or aerospace engineering, the designs are tested against a range of laboratory experiments backed up by expert judgement based on experience with the same or similar systems. Design for a repository for radioactive waste deviates from standard engineering practice in that no HLW repositories (and only a few L/ILW repositories) yet exist and, even when they do, testing their compliance to design limits will be somewhat difficult due to the timescales involved (see discussion in Alexander et al., 2003).

In this discussion so far there has been little mention of risk in its probabilistic sense. This is because risk perception is a multivariate problem, i.e. it does not rely on simple numerical considerations alone (Renn et al., 1996). Wiedemann (1997) provides a review of earlier findings which show that the public's understanding of the probability of dying from a range of "every day" causes (cancer, tuberculosis, stroke, homicide, tornadoes, road accidents) fairly reflects the numerical estimates. However, the less frequent the occurrence of a hazard, and the greater its catastrophic potential, the greater the overestimation of the associated risk – the combination of unfamiliarity and "dread" again.

For most people, risk is not thought of in terms of a probability. The more remote a risk is perceived to be, the less the degree of control which may be exercised over it. This explains why individuals are prepared to tolerate a greater degree of voluntary risk since they may perceive some associated benefit and because they feel themselves in control (e.g., smoking, road travel). Risks from radioactive waste disposal are very definitely interpreted as involuntary and without benefit.

Perception of technological risks are known to vary nationally – 62% of West Cumbrians (probably representative of the British population in general) rank radioactive waste disposal as "very or quite worrying", and a similar survey of French opinion shows 87% with this perception. The American public has a similar attitude to radioactive waste disposal (Mays and Poumadere, 1994). Conversely the French have a higher degree of confidence in the engineers and other experts managing nuclear power. Given the national benefits derived from nuclear power generation in Japan, it might be anticipated that a more favourable response would be forthcoming.

With this background, it may legitimately be asked what can be done to increase confidence in risk estimates arising from such safety assessments as H12. Clearly, satisfying regulatory concerns alone is insufficient to ensure public acceptability of waste management options. Of prime importance is the need to involve the public in the risk management process from the earliest stage (Walker et al., 1997) as consensus building requires a mutual desire to find a solution. The US DOE (1999) comparison of the action/no-action alternative is useful in this respect since it encourages recognition that there are choices to be made in waste management strategy. However, this requires that the full background to the situation be available to the participants in the process so that they are able to judge for themselves. Credibility is, of course, all important. Clear evidence of *concern* and *care* on the part of industry and *commitment* by government would increase public confidence and trust (Covello and Peters, 1996). Similarly, increased public understanding of the issues would lead to greater confidence in the process.

While safety assessments alone cannot deliver public acceptance, they do provide important information to the process, contributing to the technical background. For the non-expert technical audience, efforts must be made to present both the conceptual ideas involved in the SA and the results in a clear and open manner. For non-technical audiences, presentation of the fundamental safety messages in a simple and unambiguous manner may require alternative means such as the use of natural analogues (see, for example, West et al., 2002, Tsuboya and McKinley, 2003). It should be acknowledged that the public, as well as the regulator, must be persuaded by the argument of the SA. A full debate of the need for disposal is required and this implies that the "do-nothing" alternative also be evaluated in such a way that the general public perceives the reasonableness of the proposed solution.

7.2 Risks associated with nuclear waste disposal doses

As discussed in section 7.1, public perception of risk may not be reduced to a simple comparison of numerical risks from different sources. Nevertheless, it was also noted above that the debate about nuclear power issues in general, and radioactive waste disposal in particular, would be better facilitated with a wider background of public knowledge and greater openness on the part of the proponents of disposal options. Part of this must rely on the comparison of risks both in everyday life (voluntary and involuntary) as well as those perceived as being less familiar, such as the risk from waste disposal. This helps to set not only the context of the risk calculated in the H12 study, but also indicates the degree of protection offered by the public regulations regarding ionising radiation in Japan.

In contrast to other known cancer-inducing agents, the relationship between ionising radiation dose and health impact is well established, through the publications of the International Commission on Radiological Protection (e.g., ICRP, 1991). Although there is some debate, the *linear, no threshold* response model is broadly accepted[43] for radiological protection purposes. This means that there is no dose rate below which a detriment may not be calculated and that the response at low dose rates may be extrapolated from the effects seen at higher dose rates. The ICRP (1991) risk factors are:

- Fatal cancer in exposed individual 0.05 Sv^{-1}
- Serious hereditary defect in all generations of offspring 0.01 Sv^{-1}
- Allowance for loss of life expectance and non-fatal cancer 0.01 Sv^{-1}.

The peak dose rate calculated in the H12 Reference Case is $5x10^{-3}$ μSv a^{-1}, corresponding to an individual risk of fatal cancer of $2.5x10^{-10}$ per year. This dose rate is maintained close to its peak level for a period of time much greater than a human life span. This allows the lifetime risk of death from radiation induced cancer due to releases from the HLW repository to be estimated as $1.75x10^{-8}$ (around one in 57 million) for a person living at the time of maximum releases (assuming a 70 year lifetime).

43 The work of the ICRP is under constant review. The latest revision of risk factors was published in 1991 but application of the linear no-threshold response model is currently under discussion. There are also proposals to replace current radiation protection exposure limits based on Collective Dose (over a population) with a simpler approach based on the most exposed representative individual. Under these proposals, protecting a representative individual of the mostly highly exposed group would ensure protection of the whole population

Tab. 7.2.1: Classification of annual risks of death with examples. Data from Fritzsche (1992), as cited by Baertschi and Sumerling (1994). Risks from the H12 Reference Case as well as variant cases in the SA are indicated, together with the risk of death from fatal cancer from the natural background radiation in Japan

Classification	Examples	Risk of death [per year]	
		Central value	Range
Extremely high	Circulatory illnesses, Men > 55 years Women > 65 years	$1.7x10^{-2}$ $2.4x10^{-2}$	$> 10^{-2}$
Very high	Cancer, all types and ages (men and women) Lung & bronchial cancer, men > 45 years	$2.5x10^{-3}$ $2.2x10^{-3}$	$10^{-3} – 10^{-2}$
High	Diabetes (men and women) > 55 years Cancer of the colon (men and women) > 55 years	$6.0x10^{-4}$ $8.0x10^{-4}$	$3x10^{-4} – 10^{-3}$
Medium	Leukaemia (men and women) > 35 years Influenza, men > 55 years	$1.3x10^{-4}$ $1.5x10^{-4}$	$10^{-4} – 3x10^{-4}$
Low	Multiple sclerosis men and women) > 45 years Infectious diseases (excluding AIDS)	$3.7x10^{-5}$ $8.0x10^{-5}$	$3x10^{-5} – 10^{-4}$
	Dose from natural background radiation in Japan (900 – 1200 µSv a^{-1})		$4.5x10^{-5} – 6.0x10^{-5}$
Very low	Leukaemia (men and women) 1 – 25 years Bone cancer (men and women) all ages	$1.5x10^{-5}$ $5.0x10^{-6}$	$3x10^{-6} – 3x10^{-5}$
Negligible	Lightning strike	$5.0x10^{-7}$	$< 3x10^{-6}$
H12	**Reference Case[1]** **Fresh, reducing high pH groundwater, high flow[1]** **Fresh, reducing high pH groundwater, low flow[1]**	**$2.5x10^{-10}$** **$1.0x10^{-6}$** **$5.0x10^{-14}$**	

Note:
1: For an individual living at the time of maximum releases (ca. 1 Ma in the future).

Before this, the dose, thus annual risk, is lower. Baertschi and Sumerling (1994) cite data from Fritzsche (1992) which provided a classification of annual risks of death from a number of sources (see Table 7.2.1). By comparison it may be seen that risks calculated in the H12 project are of very low order – below negligible - by this classification. Even variants on the Reference Case which give rise to higher dose rates, and hence risks, are no greater than negligible. For comparison, the natural background radiation dose in Japan ranges from 900 to 1200 µSv a^{-1} and, within this range, the risk is in the "low" category.

Modern human society gives rise to a number of exposures to ionising radiation from man-made or artificially enhanced sources. For example, in central Europe, the combined dose from Chernobyl fallout and medical diagnoses is around 1000 µSv a^{-1} ($5x10^{-5}$ risk per year of death) – comparable to that from the Japanese background radiation. Air travel exposes passengers and crew to enhanced cosmic radiation – a ten hour intercontinental flight corresponds to around 50 μSv – a risk of death of $3.6x10^{-8}$ (Baertschi and Sumerling, 1994).

Table 7.2.2 lists some specifically Japanese risks for comparison. It is clear that although the risk of death in an earthquake is "very low" on the Fritzsche (1992) classification, it is several orders of magnitude higher than the risk associated with a radioactive waste repository as calculated in H12. The other risks due to natural phenomena listed in Table 7.2 would be classified as "negligible" or below, but the annual risk of death by tornado in Japan is still an order of magnitude higher than the estimated radiological risk for the H12 Reference Case.

Tab. 7.2.2: Annual risk of death due to natural phenomena in Japan

Examples	Risk (x 10^6) [1]	Number of deaths	Period	Reference[2]
Earthquake[3]	5.3	6623	1988 -1997	A,B
Typhoon	0.4	206	1988-1991	C
Heavy rain	0.3	156	1988-1991	C
Snow/ice/avalanche	0.13	67	1988-1991	C
Volcanic eruption	0.09	43	1988-1991	C
Hail / lightening	0.03	16	1988-1991	C
Strong winds	0.03	13	1988-1991	C
Tornado	0.004	2	1988-1991	C

Notes:
1: Assuming that Japan has a population of 125 million
2: A: Institute for Fire Safety and Disaster Preparedness (1997)
B: Okushiri Town (1996)
C: Institute for Fire Safety and Disaster Preparedness (1993)
3: Hanshin/Awaji January 1995: 6425 deaths; Hokkaido-Nanseioki (off-shore, southwest) July 1993: 198 deaths.

Tab. 7.2.3: Activities giving rise to an estimated one-in-a-million lifetime risk of fatality (Baertschi and Sumerling, 1994)

Activity	Potential Hazard
Smoking 2 cigarettes	Cancer and circulatory disease
Living with a cigarette smoker for 2 months	Cancer and circulatory disease
Driving 300 km	Accident
Cycling 50 km	Accident
Flying 4 000 km (commercial airline)	Accident
4 hour flight at 10,000 m altitude	Cancer from cosmic radiation

Although the public perceive a high risk from waste disposal, using subjective measures to increase the perceived risk because of "unfamiliarity" and "dread" (section 7.1), their "dread factor" is often based on the assumption of great catastrophic potential, i.e., that large numbers of people would be affected. Although the collective dose impact associated with the H12 case has not been calculated, the very low risks for the H12 concept indicate that rates of fatal exposure would be very low. Also, if the maximum dose is applied to a representative individual in the most exposed group (identified critical group, in SA terminology), in line with the ICRP's proposals, the collective dose estimated for the population would represent a maximum, since many affected individuals would be less exposed than the most exposed. Unfortunately, it is difficult to estimate the size of the potentially affected area in a generic study and, therefore, the size of the population involved. Even so it is unlikely that large numbers of people would be directly affected at similar concentrations to those calculated for the critical group in the study.

The "dread" factor acts to inflate the perceived risk when compared to the numerical value calculated in risk assessments (see Wiedemann, 1997) and it is interesting to compare with similar risks, many of which are undertaken voluntarily. Lifetime risks at the level of one-in-a-million (10^{-6}) from a number of normal activities are compiled in Table 7.2.3 (from Baertschi and Sumerling, 1994) to indicate the significance of this level of risk. It should be noted that for types of risk which give rise to cancer or other diseases there may be a significant latency period, whereas for other risks death is immediate. Examples would be accidental death in a car or in an earthquake compared to death by cancer (or other illnesses) due to cigarette smoking that results in reduced life expectancy rather than immediate death. This latency period may act to reduce the perceived risk associated with an activity.

Few people, even non-smokers, would assign a significant risk to smoking just 2 cigarettes, yet this is a similar level of risk to that calculated for a conservative (high flow) H12 case (exposure for 1 year).

Other safety assessments of HLW and spent fuel repositories also calculate low maximum doses to H12, implying similarly negligible risks to an individual. For example, for TILA 99, the maximum dose is around 0.1 $\mu Sv\ a^{-1}$ resulting in a risk of cancer induced fatality of around $3x10^{-7}$ (assuming a 70 year lifetime). Unlike H12, where the maximum risk would arise at around 1 Ma after repository closure, in TILA 99 the maximum risk is close to 10000 years after repository closure[44], due to the volatile radionuclides (instant release fraction) released after overpack failure. The SITE 94 assessment gave a higher maximum dose - around 10 $\mu Sv\ a^{-1}$ - but this still corresponds to an annual risk of only $5x10^{-7}$ or a lifetime risk of $3x10^{-5}$.

For comparison, Baertschi and Sumerling (1994) calculated a dose of 30 $\mu Sv\ a^{-1}$ to an individual from drinking 2 litres of average Swiss mineral water per day (0.73 m^3 per year). This is equivalent to a risk of around 10^{-6} per year. It is clear that the risk associated with ingestion of naturally occurring radionuclides in mineral water can be significantly greater than those estimated to arise from releases from a radioactive waste repository. In fact, the maximum annual dose calculated in the Kristallin-I SA is equivalent to drinking just 5 litres of average Swiss mineral water per year.

From this, it is clear that the nuclear waste industry has a great deal of work to do in explaining radioactive waste disposal to the general public, if the "dread" factor is to be sufficiently reduced to ensure public acceptance (see also comments and risk comparisons in West et al., 2001).

44 The maximum releases (and doses) for spent fuel occur relative soon after overpack failure due to the volatile nature of the radionuclides assumed to be released immediately from the surface of the spent fuel. These highly soluble nuclides, such as C-14, I-129, Cs-135 are not wholly held within the fuel and the "instant release fraction" is not dependent on dissolution of the fuel before release. However, the time at which a overpack fails for the purposes of the TILA 99 calculation is arbitrary (the overpack "disappears" at 10^4 years). In the expected evolution of the Finnish repository, no overpacks will fail before 10^6 years

8 CONCLUSIONS

F.B. Neall, P.A. Smith

The key points arising from the foregoing chapters are summarised in this section, followed by conclusions for the report as a whole.

8.1 Strategy for making the safety case

The generic nature of the host rock in the H12 assessment means that emphasis is placed very much on a strong near field performance. Within the EBS, the bentonite buffer has a key role since the overpacks are assumed to fail after 1000 years, and no further benefit is taken from their presence, apart from redox buffering.

The bentonite provides a robust colloid filter as well as chemical buffering and ensures radionuclide release by diffusion only. Radionuclide concentrations are also limited in the near field by the slow release from the waste glass matrix. This robust EBS is consistent with the strategy employed in other national programmes.

In general, the strategy for making the safety case was consistent throughout H12, although the focus on the Tono and Kamaishi test sites as the main source of geological data is rather unusual. Neither area would normally be considered as a repository site due to inadequate depth and the high degree of host rock fracturing that results in high water flow underground.

8.2 Approach to safety assessment

H12 follows a widely accepted approach to safety assessment, as described by NEA (1999b):

- Scenario development
- Consequence analysis
- Assessment of available safety margins.

Screening criteria for the selection of phenomena (FEPs) to be taken into account are tailored to the specific disposal system under consideration and the aims of the SA. In H12, the generic nature of the SA means that assumptions must be made regarding the nature of the site and it's susceptibility to processes such as volcanism, faulting, uplift etc.

The H12 overpacks are assumed to have a limited lifetime, unlike some other assessments where the safety concept includes a much longer-lived overpack. The H12 approach avoids both the problems of having to justify expectations of very long lifetime for the overpack, and how to obtain a realistic failure rate for the overpacks. As TILA 99 and SR 97 acknowledge, there is currently little or no evidence on which to estimate the probability or rate of overpack failure.

In H12, uncertainty is addressed by a combination of alternative scenarios, alternative conceptual models and parameter variations used in deterministic calculations. This allows the significance of various types and scales of uncertainty to be investigated, leading to improved system understanding, which can in turn guide where effort is best concentrated in order to increase the reliability of future assessments, and possibly reduce the level of conservatism in the analysis.

In addressing near field features and processes, H12 uses realistic data and models where these are well supported, for example by experimental results, and conservative data elsewhere. This is intended to ensure that the results provide bounding estimates of radiological impact that err on the side of pessimism (i.e. produce higher doses), while keeping the degree of conservatism to a level where results can still be used to assess the impact of various processes within the analysis.

By comparing parameter values for the near field models between various SAs, it is clear that variations in conceptual models can be more significant than data differences. For example, for diffusion through the bentonite buffer, the outer boundary condition for the bentonite is the main difference between SAs; the apparent diffusivities used are mostly consistent and most variation arises with redox sensitive nuclides, as redox affects sorption and diffusion for nuclides that can form both anionic and cationic species.

With respect to geosphere transport, the transport resistance[45] provided by the host rock is similar in H12 (for fractures with the geometric mean transmissivity) and SKB 91 (central values), and about an order of magnitude higher than the transport resistance in H3, TILA 99, Kristallin-I, and SITE 94. This difference may be attributed, at least in part, to assumptions regarding small-scale heterogeneity and, specifically, the "channelling factor", i.e. the proportion of fracture surfaces assumed to be in contact with flowing water. This is an order of magnitude higher in H12 compared to Kristallin-I and SITE 94.

The "channelling factor" also affects matrix diffusion since it influences the volume of the host rock which is available for diffusion. In H12, the higher "channelling factor" is combined with a greater matrix diffusion depth than used in most other SAs, leading to very favourable geosphere retardation compared to many other assessments. Despite this, the generic nature of the geosphere means that reliance for safety is still placed on the EBS to a great extent.

The biosphere provides a key factor in determining doses due to the dilution of water from the geosphere into a large volume of aquifer and river water from where drinking water and water for agricultural irrigation are extracted. The doses arising scale directly with the dilution factor. In H12, the discharge of groundwater into a large river from which drinking and irrigation water are taken means that dilution is more than an order of magnitude higher than to other assessments in which discharge takes place into restricted water bodies used for drinking water such as wells (e.g. SR 97, TILA 99, Kristallin-I).

8.3 Results from H12 compared to H3

The maximum dose arising in the H12 Reference Case is slightly lower than the equivalent case in H3 and the dominant nuclide is Cs-135, compared to Pd-107 in H3. The dose maximum also arises substantially earlier than in the H3 assessment (Fig. 5.1.2).

These changes arise due to substantial modifications to both the near field and geosphere models. In the near field model in H12, shared elemental solubility limits

45 A measure of the effectiveness of the geosphere provided by hydraulic parameters but excluding the effects of sorption and the depth of matrix diffusion(see section 4.6.1 and Tab. 4.6.1)

result, for example, in a decrease in the importance of Se-79 compared to H3 due to the influence of the large stable Se inventory (a decrease in the Se solubility limit of approximately two orders of magnitude, based on new experimental data, is also partly responsible). A less conservative boundary condition for diffusion in the bentonite, which depends on groundwater flux, also decreases many radionuclide releases.

The results of these changes, which tend to increase the near field performance in H12, are largely offset by other changes which result in a significantly deteriorated performance of the geosphere in H12 compared, for example, with the equivalent porous medium model of H3. In particular, the use of a fracture network model and data for fracture transmissivities and geometries based on measured properties has lead to a rather more realistic, if not over-pessimistic, assessment of geosphere performance, despite the still generic nature of much of the data[46].

8.4 Safety assessment results in an international perspective

Overall, the calculated dose maxima arising in the H12 Reference Case are of a similar magnitude to those of other recent SAs, such as TILA 99 and Kristallin-I. The earlier appearance of "significant" calculated doses (> 10^{-2} μSv a^{-1}) and the earlier dose maximum in H12, compared to H3, are also more similar to Kristallin-I than H3. In particular, the attenuation of radionuclide releases from the EBS that takes place in the geosphere shows H12 to be more similar in performance to other SAs than H3, which now appears in some ways to have been somewhat simplistic and non-conservative.

The identification of key factors of disposal systems that provide safety has been an outcome of SAs such as H12 carried out in recent years, as a result of a combination of sensitivity analyses and reasoned arguments. The studies identify similar key safety factors for the geosphere, irrespective of waste type. In particular,

- Protection of the bentonite from erosion or alteration, which depends on low groundwater flow around the near field and appropriate groundwater chemistry
- Low transmissivity transport pathways in the geosphere
- Conditions favourable to geosphere retardation processes

were identified as highly important in all cases. Stable physical and chemical conditions in the geosphere were also important for the longevity of the engineered barrier. H12, like other SAs considered above, recognised that further site-specific information will be required to take full advantage of the safety potential of the geosphere as a transport barrier.

Key safety factors for the near field were more varied, depending on the waste type. For example, SAs addressing spent fuel disposal must consider the influence of the nuclides assumed to be released instantaneously on failure of the overpack, which results in the high importance attached to overpack lifetime and failure rate. Since these components, in particular C-14 and I-129, are largely lost in reprocessing, and there is, in any case, no instantaneous release from vitrified HLW, the importance of the overpack is significantly less for SAs such as H12 and Kristallin-I which consider this waste form.

46 Due to the general lack of more appropriate, deep geological data, much of the data used came from the Tono and Kamaishi test sites, (shallow and highly fractured with high groundwater flow rates)

However, in all assessments, the fine pore structure of bentonite that ensures colloid filtration and diffusion-dominated transport in the EBS, is identified as highly important.

Both natural analogues and laboratory and field experimental programmes have been used in H12 to support the expected performance of the key safety factors. For example, analyses of degradation of volcanic glasses in various environments have been used to support the contention of low degradation rates for the borosilicate glass waste form, and large-scale vibration experiments on compacted bentonite, with an embedded steel mass to simulate a waste package, have been carried out to examine the behaviour of the EBS during an earthquake. In addition, a number of uranium ore deposits have been used as qualitative support for the expected performance of the total disposal system over very long timescales. One example, the Tono uranium deposit in Japan, has provided invaluable information on specifically Japanese conditions. H12 also draws on support from international analogue studies which have been carried out over the last 20 years or so, such as those at Poços de Caldas, Alligator Rivers and Oklo.

8.5 Public perception of the risks associated with radioactive waste disposal

Public perception of radiation and, in particular activities involving radioactive materials, has undergone a major change during the 20th century from generally favourable to extremely wary and suspicious. As noted above, much of this change has been ascribed to the accidents involving nuclear facilities.

Radiation routinely scores very highly in terms of perceived hazard compared to other potential risks such as those associated with transport, industrial and chemical installations, etc. This may, to a large extent, be attributed to a combination of "dread" and lack of "familiarity". The main factors influencing concern about radiation hazards seems to be that its action is unseen and not readily detectable, and has the potential to cause harm over long timescales and to be wholly outside the control of the individual. In addition, encounters with the concepts of radiation, nuclear power and radioactive waste disposal tend to be via the media, where the need for a "good story" may lead to less than balanced coverage.

Greater public understanding of the issues involved in waste disposal could help to promote greater acceptance of the need for disposal and the potential options. However, building public confidence and trust in the organisations responsible for implementing and regulating radioactive waste disposal is fundamental to its acceptance in democratic countries.

8.6 Significance of nuclear waste disposal doses

Public perception of different risks is too complex to allow a simple comparison of numerical risks from different sources to critics of radioactive waste disposal. Nevertheless, the debate about radioactive waste disposal can benefit from these types of comparison if an adequate explanation of the background is also available – helping to reduce both the "unfamiliarity" and the "dread" associated with such waste. Part of this must rely on the comparison of risks encountered in everyday life (voluntary and involuntary) with those perceived as being less familiar, such as the risk from waste disposal. This helps not only to provide an appreciation of the significance of the risk calculated in the H12 study, but also the degree of protection offered by the public regulations regarding ionising radiation in Japan.

The peak dose rate calculated in the H12 Reference Case is $5x10^{-3}$ µSv a^{-1} compared to the natural background radiation dose in Japan which ranges from 900 to 1200 µSv a^{-1}. The corresponding risk to an individual of fatal cancer is estimated to be $2.5x10^{-10}$ per year for the H12 calculated maximum dose. This means that the lifetime risk of death from radiation induced cancer due to releases from the proposed HLW repository is around $2x10^{-8}$ (assuming a 70 year lifetime) for a person living at the time of maximum releases, approximately 1 Ma in the future. Such a lifetime risk is a good deal lower than the annual risk of death by being struck by lightening ($3x10^{-8}$ per year) or being killed in an earthquake in Japan ($5x10^{-6}$ per year).

8.7 Overall conclusions

Comparing the broad SA steps and model assumptions used, H12 is seen to be consistent in many respects to assessments carried out in other countries. Some assumptions in H12 are somewhat arbitrary reflecting the generic stage of the Japanese programme. Safety assessment assumptions are likely to become better founded in future exercises, due to the development of better system understanding and SA tools, provision of additional relevant (e.g. site-specific) data, definition of a final repository design and from experience gained in H12 and its review.

Comparing this report to earlier comparison exercises such as that carried out for Kristallin-I (Neall, 1994), it is clear that there is still a consensus that well characterised engineered barriers allow more confident modelling of the near field than is the case for the more heterogeneous geosphere, at least in the case of fractured hard rocks. In the case of a more homogeneous geosphere, such as the Opalinus Clay in Switzerland, it is likely that more reliance can be placed on the behaviour of the geosphere (Nagra, 2002). Even in cases such as SR 97 and, particularly, TILA 99, which use site-specific data, there is greater reliance on a very long-lived overpack than on more precise determination of the performance of the geosphere. It will be interesting to see whether, as more programmes move to the stage of characterising a candidate site, the need to optimise the EBS for engineering feasibility and cost, as well as safety, will result in greater emphasis on the natural barrier.

As noted previously by Neall (1994), there is surprisingly good agreement in the maximum doses predicted by a number of assessments; all values lie between two to four orders of magnitude below the regulatory limit, despite a variety of concepts and models which give different weightings to the role of different barriers. This is a reflection of the procedure whereby benefit is taken for favourable processes until a "sufficient" level of safety has been achieved. Other processes become "Reserve FEPs" - to be included in the analysis if required. This approach may appear dubious to those outside the waste disposal community and efforts must be made to ensure that the corollary - that all relevant detrimental processes are included in the analysis through the FEP lists and scenario development - is adequately explained. Furthermore, no waste disposal programme has sufficient resources to investigate all potentially favourable processes to the level required for their incorporation in models for quantitative analysis, and this rather pragmatic approach allows priorities to be set and resources concentrated on examining the most important beneficial processes.

Finally, it is important to recognise that, despite the generic nature of the H12 safety assessment, it contains the clear message that it is certainly feasible to site and construct a HLW repository in Japan which will remain safe and impose negligible risk on the Japanese population.

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Appendix A – Data used to generate Figure 3.2.1

P.A. Smith

i) Data from H12 – horizontal emplacement

The vitrified waste takes the form a solid circular cylinder of diameter 0.43 m and length 1.34 m (see Section 3.2 in H12 Supporting Report 3). Its volume is:

$$\frac{\pi}{4} \times 0.43^2 \times 1.34 = 0.20\,\mathrm{m}^3$$

The steel overpack takes the form of a hollow circular cylinder. Its external diameter is 0.82 m and external length is 1.73 m, and it has a wall thickness of 0.19 m (see Fig. 5.3.1-1 in H12 Supporting Report 3). The volume of steel not including the overpack ends is:

$$\frac{\pi}{4} \times \left(0.82^2 - (0.82 - 2 \times 0.19)^2\right) \times \left(1.73 - 2 \times 0.19\right) = 0.51\,m^3$$

The total volume of steel in the two overpack ends is:

$$\frac{\pi}{2} \times 0.82^2 \times 0.19 = 0.20\,\mathrm{m}^3$$

Giving an overall volume of steel of 0.71 m^3

Sand / bentonite fills the space surrounding the overpack, out to the tunnel walls. The tunnels have a diameter of 2.22 m (see Fig. 5.3.1-1 in H12 Supporting Report 3 and the waste package pitch is assumed to be 3.13 m (see Section 6.1.2.4.2 in H12 Supporting Report 3). The volume of sand / bentonite is:

$$\frac{\pi}{4}\left(2.22^2 \times 3.13 - 0.82^2 \times 1.73\right) = 11.20\,m^3$$

The total volume of material per waste package in the H12 horizontal emplacement concept is 0.20 m3 + 0.71 m3 + 11.20 m3 = 12.11 m^3.

ii) Data from H12 – vertical emplacement

The volumes of vitrified waste and steel are the same as in (i), above. The waste packages are, however, placed in deposition holes of diameter 2.22 m and depth 4.13 m, drilled into the floor of repository tunnels (see Fig. 4 in H12 Supporting Report 2). The volume of sand / bentonite buffer used to fill the deposition holes is

$$\frac{\pi}{4} \times \left(2.22^2 \times 4.13\right) - \left(0.82^2 \times 1.73\right) = 14.82\,\mathrm{m}^3$$

The tunnels have a width and height of 5.0 m, with semi-circular rooves. The deposition holes have a spacing of 4.44 m. The volume of sand / bentonite used to backfill the tunnels (per deposition hole) is:

$$4.44\times 5\times 2.5+4.44\times \frac{\pi}{8}\times 5^2 = 99.09\ \mathrm{m}^3$$

iii) Data from TVO 92

The composite copper / steel overpack has an external diameter is 0.8 m and a length of 4.5 m. The thickness of copper is 0.06 m and the thickness of steel is 0.055 m (see Chapter 3 TVO 92). The volume of copper not including the overpack ends is:

$$\frac{\pi}{4}\times\left(0.8^2-(0.8-2\times 0.06)^2\right)\times(4.5-2\times 0.06)=0.61\ \mathrm{m}^3$$

The total volume of copper in the two overpack ends is:

$$\frac{\pi}{2}\times 0.8^2\times 0.06=0.060\ \mathrm{m}^3$$

Giving an overall volume of copper of 0.67 m^3.

The volume of steel not including the overpack ends is:

$$\frac{\pi}{4}\times\left((0.8-2\times 0.06)^2-(0.8-2\times(0.06+0.055))^2\right)\times(4.5-2\times(0.06+0.055))=0.46\ \mathrm{m}^3$$

The total volume of steel in the two overpack ends is:

$$\frac{\pi}{2}\times(0.8-2\times 0.06)^2\times 0.055=0.040\ \mathrm{m}^3$$

Giving an overall volume of steel of 0.50 m^3.

The inner volume of the overpack, containing the spent fuel, other structural parts of the fuel assemblies and quartz fill, is:

$$\frac{\pi}{4}\times 0.8^2\times 4.5-0.67-0.50=1.09\ \mathrm{m}^3$$

The waste packages are placed in deposition holes of diameter 1.5 m, the bottom 6.5 m of which is filled with compacted bentonite (Fig. 3.4 in TVO 92). The volume of compacted bentonite used per waste package is:

$$\frac{\pi}{4}\times\left(1.5^2\times 6.5\right)-\left(0.8^2\times 4.5\right)=9.22\ \mathrm{m}^3$$

Sand / bentonite fills the top 1 m of the deposition holes, as well as the repository tunnels. The volume of sand / bentonite in the deposition holes is:

$$\frac{\pi}{4}\times 1.5^2\times 1.0=1.77\ \mathrm{m}^3$$

The repository tunnels have deposition holes drilled into the floor, with a separation of 6 m. The tunnels have a width of 3.3 m and a height of 4.6 m and have semi-circular rooves. The volume of sand / bentonite in the tunnels (per deposition hole) is:

$$6\times 3.3\times(4.6-3.3/2)+6\times\frac{\pi}{8}\times 3.3^2 = 84.07\ \mathrm{m}^3$$

giving a total volume of sand / bentonite of 85.84 m^3.

The total volume of material per waste package in the TVO 92 concept is 0.67 m^3 + 0.50 m^3 + 1.09 m^3 + 9.22 + 85.84 m^3 = 97.32 m^3.

Appendix B – Models and data used to generate Figures 3.2.2 and & 3.2.3

i) Percentage decay before overpack breaching

Before overpack breaching, neglecting radioactive ingrowth, the percentage decay P_1 at the time of overpack breaching t_b [a] is given by:

$$P_1 = 100\left(1 - e^{-\lambda t_b}\right)$$

where λ [a^{-1}] is the decay rate, which, in Figure 3.2.2, is based on the half lives given in Table 5.3.1-2 of H12 Supporting Report 3. The time of overpack breaching in the H12 concept is 1000 years (see, for example, Section 5.3.1.1.1 in H12 Supporting Report 3). A breaching time of one million years is assumed for the TILA 99 concept in generating Figure 3.2.3. In TILA 99, the best estimate is that the overpack will remain intact for " ... more than one million years" (see Ch. 13 in POSIVA 1999)

ii) Percentage decay of releases from the waste form during near field transport

A simple, one-dimensional steady-state model is used to illustrate the percentage decay of releases from the waste form during near field transport, following overpack breaching.

The steady-state equation governing radial diffusion through the bentonite surrounding the overpack is:

$$0 = \frac{1}{r}\frac{d}{dr}\left(r\frac{dC}{dr}\right) - \frac{\lambda R_b}{D_b}C$$

where C [mol. m^{-3}] is radionuclide concentration, D_b [$m^2\ a^{-1}$] is the pore diffusion coefficient and:

$$R_b = 1 + \frac{\rho_b K_b}{\varepsilon_b}$$

ε_b is the bentonite porosity, ρ_b [kg m^{-3}] is its dry density and K_b [$m^3\ kg^{-1}$] is the sorption coefficient of the radionuclide under consideration.

Solving this governing equation for a fixed concentration at the inner boundary of the bentonite ($r = r_a$ [m]) and assuming $C \to 0$ as $r \to \infty$, the percentage decay P_2 during transport to a distance $r = r_b$ [m] is given by:

$$P_2 = 100\left(1 - \frac{r_b K_1(r_b')}{r_a K_1(r_a')}\right)$$

where

$$r_a' = r_a\sqrt{\frac{\lambda R_b}{D_b}},\ r_b' = r_b\sqrt{\frac{\lambda R_b}{D_b}},\ \text{and}$$

$K_0(r')$ is a modified Bessel functions of order zero.

Data used to generate Figure 3.2.2 are taken from H12 Supporting Report 3. In particular, they are taken from Figure 5.3.1-1 in for r_a, r_b and ε_b, Table 5.3.1-8 for D_b [47] and Table 5.3.1-9 for K_b.

Data used to generate Figure 3.2.3 are taken from POSIVA (1999). In particular, they are taken from Table 11-7 for r_a and r_b, Table 11-4 for K_b and Table 11-5 for ε_b and D_b.

Both figures use the same value for ρ_b, taken from Figure 5.3.1-1 in H12 Supporting Report 3.

iii) Percentage decay of near-field releases during geosphere transport

A simple, one-dimensional steady-state model is also used to illustrate the percentage decay of near field releases during geosphere transport.

The steady-state equation governing transport by advection / dispersion through a parallel walled channel of aperture $2b$ [m] conveying flowing water in the z-direction through the geosphere is:

$$0 = -u\frac{\partial C_f}{\partial z} + a_L u \frac{\partial^2 C_f}{\partial z^2} + \frac{\varepsilon_p D_p}{b}\left.\frac{\partial C_p}{\partial y}\right|_{y=b} - \lambda C_f$$

where C_f [mol. m^{-3}] is radionuclide concentration in the channel and C_p [mol. m^{-3}] is the radionuclide concentration in the surrounding rock matrix.

u [m a^{-1}] is the water velocity, given by:

$$u = \frac{Ti}{2b}$$

where T [m^2 a^{-1}] is the transmissivity of the channel and i is the hydraulic gradient.

a_L is the longitudinal dispersion length, given by:

$$a_L = \frac{L}{Pe}$$

where Pe is the Peclet Number and L [m] is the length of the channel.

The steady-state equation governing matrix diffusion is:

$$0 = \frac{d^2 C_p}{dy^2} - \frac{\lambda R_p}{D_p} C_p$$

D_p [m^2 a^{-1}] is the pore diffusion coefficient in the matrix and:

$$R_p = 1 + \frac{\rho_p K_p}{\varepsilon_p}$$

ε_p is the matrix porosity, P_p [kg m^{-3}] is its dry density and is K_p [m^3 kg^{-1}] is the sorption coefficient of the radionuclide under consideration.

Solving the governing equations for a fixed concentration at the upper boundary of the channel ($z = 0$) and assuming $C_f \rightarrow 0$ as $r \rightarrow \infty$ and a maximum depth for matrix diffusion of y_p [m], the percentage decay P_3 during transport to a distance $z = L$ is given by:

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47 The tables in H12 Supporting Report 3 and POSIVA (1999) give effective diffusion coefficients. These are divided by the bentonite porosity in order to obtain the required pore diffusion coefficients

$$P_3 = 100\left(1 - \exp\left\{\frac{Pe}{2}\left[1 - \sqrt{1 + \frac{8b}{Pe}\frac{LR_{eff}}{Ti}\lambda}\right]\right\}\right)$$

where

$$R_{eff} = 1 + \frac{\varepsilon_p R_p}{b'}\tanh\left(y'_p\right)$$

$$b' = b\sqrt{\frac{\lambda R_p}{D_p}} \text{ and } y'_p = y_p\sqrt{\frac{\lambda R_p}{D_p}}$$

For cases where longitudinal dispersion is negligible (as assumed in TILA 99), $Pe \to \infty$ and:

$$P_3 \to 100\left(1 - \exp\left\{-\frac{2bLR_{eff}}{Ti}\lambda\right\}\right)$$

The fracture aperture, 2*b*, which is an insensitive parameter, is set arbitrarily to 1 mm.

Other data used to generate Figure 3.2.2 are taken from Table 5.3.2-2 in H12 Supporting Report 3 for K_p and Table 5.3.2-4 in H12 Supporting Report 3 for other geosphere parameters. T is set to 160 $m^2\ a^{-1}$ ($2 \times 10^{-7}\ m^2\ s^{-1}$) for both the host rock and the fault. In the case of the host rock, this represents the upper (pessimistic) end of the range of fracture transmissivities given in Table 5.3.2-4 in H12 Supporting Report 3, multiplied by a factor of 2 to take into account the effects of channelling within a fracture.

Other data to generate Figure 3.2.3 are taken from Tables 11-4, 11-5, 11-7, 11-9 and 11-10 and 11-19 [48] in POSIVA (1999). Conservative, non-saline, reducing conditions are assumed. The entire matrix out to a distance y_p = 0.1 m is assigned uniform properties. For simplicity (unlike the TILA 99 geosphere model), no distinction is made between the first 1 cm of wallrock, which may be altered, and the remainder.

48 The quantity WL/Q in Table 11-19 of POSIVA (1999) is equivalent to L/Ti in the notation of the present report. A value of 5×10^4 is assumed for the present study