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Issues relating to safety standards on the geological disposal of radioactive waste

Proceedings of a specialists meeting held in Vienna, 18–22 June 2001

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FOREWORD

Within the International Atomic Energy Agency (IAEA) focus is currently being placed on establishing safety standards for the geological disposal of radioactive waste. This is a challenging task and a Specialists Meeting was held from 18 to 22 June 2001 with the intention of providing a mechanism for promoting discussion on some of the associated scientific and technical issues and as a means of developing the consensus needed for establishing the standards. The meeting used, as its basis, a number of position papers developed in recent years with the help of a subgroup of the Waste Safety Standards Committee (WASSC), the subgroup on Principles and Criteria for Radioactive Waste Disposal, together with selected relevant regional and national papers.

This report contains the summaries of the various sessions of the Specialists Meeting together with the conclusions drawn relevant to the establishment of standards.

The meeting was organized as part of the programme of the IAEA's Division of Radiation and Waste Safety. The responsible officer was G. Linsley.

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1. INTRODUCTION

1.1. A short history of the development of IAEA waste safety standards

The IAEA's involvement in the management of radioactive waste started soon after its creation in 1957. At that time, the disposal of radioactive waste in the sea was an option being favoured by countries developing nuclear power, and in 1961 the IAEA published Safety Series No. 5, which was concerned with establishing appropriate safety procedures and practices for the disposal of radioactive waste in the sea. This was followed a few years later by international guidance on radioactive disposal in the ground (Safety Series No. 15, 1965).

By the late 1970s, it had become clear that underground disposal was the internationally accepted approach for most types of solid radioactive waste. In 1977 the IAEA outlined a programme for the production of a set of guideline documents on the subject. A review committee was established to oversee the production of the documents. This committee, the Technical Review Committee on Underground Disposal of Radioactive Waste, came into existence in 1978 and continued its work until 1988. During this time, it approved the issuance of a comprehensive collection of Safety Series documents on the subject of underground disposal. Some of them established international norms for the planning and establishment of underground waste repositories.

Towards the end of the 1980s the issue of radioactive waste and their management was becoming increasingly important in the political sphere. It was seen as one of the technically unresolved issues of nuclear power. The IAEA responded by establishing a high profile family of safety standards, the Radioactive Waste Safety Standards (RADWASS). By this means the IAEA intended to draw attention to the fact that there were already in existence well-established procedures for the safe management of radioactive waste. The programme was intended to establish an ordered structure for the safety documents on waste management and to ensure comprehensive coverage of all relevant subject areas.

The initial concept of RADWASS was developed in 1988. The structure, content and scope of the programme was elaborated by international experts in 1990 and work on the programme started in 1991.

In 1996, the RADWASS programme was amended to broaden its scope by giving emphasis to discharges and environmental restoration and to reduce the number of planned documents by combining several of the previously planned Safety Guides. RADWASS documents are categorized under four subject areas — discharges, pre-disposal, disposal, and environmental restoration.

A number of bodies have been set up for the preparation and review of IAEA safety standards. The Commission on Safety Standards (CSS) is a standing body of senior government officials holding national responsibilities for establishing standards and other regulatory documents relevant to nuclear, radiation, waste and transport safety. It has a special overview role with regard to the IAEA's safety standards and provides advice to the Director General on the overall programme related to safety standards. In addition, four committees have been set up: the Nuclear Safety Standards Committee (NUSSC), the Radiation Safety Standards Committee (RASSC), the Waste Safety Standards Committee (WASSC), and the Transport Safety Standards Committee (TRANSSC). They are standing bodies of senior regulatory officials with technical expertise in nuclear safety, radiation safety, radioactive waste safety, and radioactive materials transport safety, respectively. They provide advice to the Secretariat on the overall programmes — and have the primary roles in the development and revision of the safety standards in their respective areas of safety.

1.2. Programme status of RADWASS

The leading RADWASS document in the category of Safety Fundamentals — The Principles of Radioactive Waste Management — was issued as Safety Series No. 111-F in 1995. This document establishes the basic principles and concepts for safe radioactive waste management.

In many areas of radioactive waste management, there is experience of the successful and safe operation of facilities; for example, in the areas of waste processing and storage, near-surface disposal, and gaseous and liquid discharge. In other areas, notably geological disposal and environmental restoration, little or no experience has yet been gained. Safety concepts and methodologies are still developing in these areas and the RADWASS programme has to reflect this fact — it is not possible to be definitive on all relevant safety issues at the present time.

To date, one Safety Fundamentals publication, three Safety Requirements and eight Safety Guides have been issued by the IAEA. They are:

- Safety Fundamentals: The Principles of Radioactive Waste Management (1995).
- Safety Requirements: Legal and Governmental Infrastructure for Nuclear, Radiation, Waste and Transport Safety (2000), Pre-Disposal Management of Radioactive Waste, including Decommissioning (2000); Near Surface Disposal of Radioactive Waste (1999).
- Safety Guides: Classification of Radioactive Waste (1994); Regulatory Control of Radioactive Discharges to the Environment (2000); Decommissioning of Nuclear Power Plants and Research Reactors (1999); Decommissioning of Medical, Industrial and Research Facilities (1999); Decommissioning of Nuclear Fuel Cycle Facilities (2001); Siting of Near Surface Disposal Facilities (1994); Safety Assessment for Near Surface Disposal of Radioactive Waste (1999); Siting of Geological Disposal Facilities (1994).

1.3. Standards for the disposal of radioactive waste

Over the last two or three decades, experience in the disposal of low and intermediate level waste in near surface repositories has been gained in many countries; however, to date, no deep geological repositories for high level waste are in operation. As a reflection of this situation, new safety standards have been developed for near surface disposal but not yet for geological disposal.

A document at the level of a Safety Requirement on near surface disposal (WS-R-1, Near Surface Disposal of Radioactive Waste) was issued in June 1999 after its approval by the IAEA's Board of Governors in March 1999. It establishes the essential radiological criteria governing this practice and the basic safety considerations for all stages of the development, operation and closure of the repository. It is supported by two Safety Guides, one on siting (SS 111-G-3.1, Siting of Near Surface Disposal Facilities), published in 1994, and the other on safety assessment (WS-G-1.1, Safety Assessment for Near Surface Disposal of Radioactive Waste), which was issued in July 1999.

The IAEA's existing safety guidance on the underground disposal of high level radioactive waste is contained in Safety Series No. 99, *Safety Principles and Technical Criteria for the Underground Disposal of High Level Radioactive Wastes*, published in 1989. However, this is an area in which safety concepts are still developing and the ICRP, the Nuclear Energy Agency (NEA) of the Organisation for Economic Co-operation and Development (OECD) and the IAEA are contributing to the achievement of consensus by supporting international expert working groups on the subject.

Extensive discussion on the safety of geological disposal has recently taken place in WASSC, particularly through the Committee's Subgroup on Principles and Criteria. The types of waste that are usually considered for deep geological disposal are characterized by their high activity (they may be heat producing) and by their content of long lived radionuclides. This requires that they be isolated and contained for very long times. Several new areas of concern were identified related to the characteristics of these wastes. They include: the definition and contents of a safety case suitable for licensing of a geological repository, the safety implications of providing for nuclear safeguards at a repository and of making provisions to allow waste retrieval, the treatment of potential human intrusion into a repository in safety assessment, the specification of alternative indicators of safety, the reference biosphere concept for use in the assessment of long term behaviour of the repository, and provisions for surveillance and control in the post closure period of the repository.

1.4. The Specialists Meeting

Focus is currently being placed on establishing standards for the "*geological disposal of radioactive waste*". This is a challenging task and the Specialists Meeting was intended as a mechanism for promoting discussion on some of the associated scientific and technical issues and as a means of developing the consensus needed for establishing the standards. The meeting used, as its basis, a number of position papers developed in recent years with the help of a subgroup of the Waste Safety Standards Committee (WASSC), the subgroup on Principles and Criteria for Radioactive Waste Disposal, together with selected relevant regional and national papers.

The sessions of the Specialists Meeting addressed the following topics:

Common framework for radioactive waste disposal

Different disposal options are adopted for different waste types and while the practice being followed in countries is generally consistent it would be useful to have a coherent and technically defensible strategy which covers the disposal of all major waste types.

Making the safety case — demonstrating compliance

The safety case for a geological repository may have to satisfy/convince persons from a variety of backgrounds, not all technical. While the core of the safety case should be based upon a formal performance assessment, other arguments may also need to be brought to bear, for example, the perspective of natural analogues, arguments on the retrievable nature of the waste.

Safety indicators

While radiation dose and risk are the main indicators of safety currently used in the safety assessment of radioactive waste disposal it has been thought desirable to supplement, and in some circumstances replace, dose and risk by other indicators, such as environmental concentrations and biospheric fluxes, which may have a lesser amount of predictive uncertainty associated with them.

Reference critical groups and biospheres

In the far future it will be impossible to predict with any reliability the nature of the biosphere overlying the waste repository and of the critical group of persons occupying it. Reference biospheres and critical groups are therefore developed as appropriate standards or references for use in converting the results of geosphere transport predictions to dose.

Human intrusion

Assessing the implications of human intrusion into a geological repository and establishing criteria for deciding on the acceptability of the disposal system in this context has been an issue of debate for many years.

Reversibility and retrievability

The recent focus on reversibility and retrievability is, in part, a reflection on the need to reassure stakeholders that actions taken now can be rescinded. Is this now a condition for any type of radioactive waste disposal? It is clear that it should not have an adverse effect on safety.

Monitoring and institutional control

Although geological repositories will be designed to provide passive safety after closure, mainly for public reassurance purposes, it is likely that some institutional controls including monitoring of the surface environment of the repository will be retained. Surveillance and monitoring may also be required for nuclear safeguards purposes.

2. SUMMARIES AND CONCLUSIONS OF THE SPECIALISTS MEETING

The presentations by the chairmen of each session, modified to take account of the subsequent discussion, form the basis for the summaries and conclusions.

2.1. Common framework for radioactive waste disposal

The Rapporteur of this session (I. Barraclough) in the absence of the Chairman (J. Cooper) observed that the talks in the session serve as background material for the other sessions in this meeting.

The disposal of many types of waste, particularly those from the ‘back end’ of the nuclear industry, can be accommodated within the current system of principles and criteria but there are some problematic issues — mining and milling waste are one example. For mining and milling waste, assurance of long term safety is often dependent upon institutional control being maintained virtually in perpetuity. The work on the common framework for the disposal of all types of radioactive waste has highlighted (if not necessarily resolved) some key issues, such as the meaning of providing the same level of protection in the future as provided at the present time and the role of institutional control in long term safety.

A conclusion was that while a common framework does appear to be possible, perhaps drawing on some of the ideas presented in ICRP Publication 81, the outcome of the IAEA programme in this area should not be such that it could adversely affect existing waste management programmes in which the waste is currently being disposed of in a satisfactory manner.

2.2. Making the safety case — demonstrating compliance

The Chairman of this session (W. Reamer) chose to summarize the presentations and discussion by listing the elements of the safety case which should be included in a standard.

Safety case

Firstly it is important to define the term "safety case" for the purpose of the standard.

The following items should be covered in the safety case:

(a) the implementer should be required to clearly document his/her arguments and supporting information

(b) the safety case should be appropriately constructed bearing in mind the need to convince different audiences

(c) a well structured approach should be used which allows for the iterative development and improvement of the safety case as new information becomes available

(d) it should be built on the safety assessment and describe the results of the safety assessment

(e) it should describe the overall safety strategy and the safety functions of the various barriers

(f) it can be both quantitative and qualitative

(g) uncertainty should be treated explicitly and the implications for repository performance should be made clear

(h) it should make use of multiple lines of evidence or reasoning

(i) it should demonstrate safe repository performance and compliance with regulatory requirements

(j) it should address safety indicators and how they are used

(k) it should provide sufficient confidence in safety such that consent can be obtained to move to the next step in the approval process

(l) it should describe planned work for the future

(m) it should address the issues of human intrusion, retrievability and multiple barriers (i.e., if and how the safety case relies on more than one barrier)

(n) it should address the use of deterministic and probabilistic assessment approaches.

Safety assessment

There are linked requirements to be addressed under "safety assessment":

(a) first a clear definition is required

(b) the safety case should involve the use of an iterative process

(c) it should use integrated performance assessment modelling

(d) it should cover both pre-closure and post-closure periods

(e) the safety assessment should provide an estimate of expected repository performance rather than a prediction of the future

(f) it should provide a quantitative measurement of repository performance in dose or risk terms

(g) it should verify compliance with regulatory requirements

(h) it should explicitly account for uncertainty, assumptions and judgement

(i) it should treat all views and address the issue of bias

(j) the results of the assessment should be traceable and the process transparent.

Decision making process

Finally, the safety case and the safety assessment should relate to the decision making process. It should be made clear that the context for both the safety case and the safety assessment is a stepwise decision making process. The nature of the safety case and safety assessment should reflect the particular decision at hand and the decision maker. They should provide a clear indication of the level of confidence which can be attached to the repository performance and the implementing organization. Context for the results of the assessments should be given in terms of the levels of safety provided and the time-scales being discussed.

International standards

The Chairman's summary, modified taking into account the final discussion as set out above, provides a good basis for the consideration of this matter in the standards.

2.3. Safety indicators

The Chairman of this session (Y. Kawakami) suggested that the following recommendations should be considered in the context of developing international standards:

(a) explicit and quantitative criteria for non-dose/risk indicators are not recommended for inclusion in international standards

(b) non-dose/risk indicators should be included in safety cases in a manner that is appropriate to the assessment context and time-scales

(c) non-dose/risk indicators other than concentration and flux should also be considered

(d) there is a need to have a clear definition of terms in this area, e.g., safety indicator, performance indicator, natural indicator.

On the last point, the discussion showed that there are different interpretations of the meaning of "safety indicator" and also that some participants favoured retaining single measures, i.e., dose and /or risk, as the target for safety assessments. They considered that introducing additional indicators on the same level would cause confusion rather than provide additional confidence. It is noted that an IAEA publication (IAEA-TECDOC-767, Safety Indicators in Different Time Frames for the Safety Assessment of Underground Radioactive Waste Repositories, 1994) had made proposals as to the time periods in which different types of safety indicators might be applicable.

International standards

In relation to a position on safety indicators for inclusion in the international standards, it seems that the general concept of non-dose/risk indicators of safety applied in appropriate time-scales is accepted but that the alternatives to dose and risk have not yet been developed to a quantitative level. For the present the standards can only include general statements in relation to the use of non-dose/risk safety indicators.

2.4. Reference critical groups and biospheres

The Chairman of the session (P. Carboneras) summarized the current status of international work on reference biospheres and indicated areas where further work needs to be done.

He noted that there is growing unanimity on the approaches to be used in the treatment of the biosphere in safety assessments. The approaches are based on scientific considerations but take due account of practical realities. The international programmes of IAEA and NEA have helped to bring about this coherence and the guidance recently issued by the ICRP has also played a part. As a result of this work there is now a set of example reference biospheres and a common methodology to define scenarios. The aim should be to keep the approach as simple as possible without being overly simplistic.

There is still room for improvement through: development of more examples; the improved availability of data; improvement of the approach for dealing with the geosphere-biosphere interface; improved treatment of sensitivity and uncertainty; improved understanding of the relationship between the source term and the reference biospheres; consolidating the definition of critical groups and further consideration of the value of environmental change modelling.

More experience is needed through the application of the reference biosphere methodology in national and international programmes. There is a need to learn from these experiences to improve the methodology and to learn how best to judge, use and present the results.

Finally, there is an urgent need for the results of the international BIOMASS programme and the associated national experiences to be summarized and documented. This may serve as an input to the development of relevant parts of geological disposal safety standards.

International standards

In summary, although significant progress has been made in recent years towards reaching a common internationally accepted method of treating the biosphere in relation to the safety assessment of geological repositories, in relation to the international standards, the conclusion seems to be that more experience, especially at the national level, in the application of the concepts of reference critical groups and biospheres, is needed before specific recommendations regarding an approach can be included in international standards.

2.5. Human intrusion

The Chairman of this session (A. Hooper) summarized the presentations and discussions under two headings: “Issues raised” and “Areas of agreement”.

Issues raised

The discussions following the presentation on the guidance from ICRP raised the issue of who is to be protected from harm in the event of an intrusion into a repository — the people living around the site, the intruder and/or “local intruders” (the distinction may not exist in the case of a near surface repository). Is there a difference if the intrusion is intentional or unintentional and between whether it is malicious or naïve? How should those potentially affected by the intrusion be protected and what is the relevance of recommended dose criteria for making judgements in this context? An earlier session prompted the question as to whether it is possible and desirable to have a common approach for treating human intrusion for near surface and geological disposal.

Areas of agreement

Perspective should be retained in discussions on this topic; it should be kept in mind that one of the main purposes of appropriately siting (in areas with little resource potential) and locating (deep) a geological repository and the planned maintenance of records and possible use of markers is to reduce the likelihood of intrusion into the repository.

The disposal system should be resilient enough to continue to provide protection for the local community even after the disruption of barriers.

It is apparent that there are differences between geological disposal compared with near-and on-surface disposal in the modes of intrusion and in the probability of intrusion. This means that commonality of approach for all disposal systems can only be at the level of principle.

ICRP 81 provides a useful basis for the development of more specific international guidance in this area but the distinction between intentional and unintentional intrusion may not be helpful for a local community and there is the possibility of confusion if ICRP guidance were to be interpreted as presenting “dual standards”. It is certainly necessary to be clearer on what are reasonable measures to avoid intrusion doses, for example, disassembling fuel bundles to achieve this purpose would involve unacceptable and actual occupational radiation doses.

There is value in establishing a commonly agreed methodology for defining scenarios for assessing human intrusion, although variations in national and local circumstances may contra-indicate fixing internationally agreed scenarios.

In the context of assessments of the consequences of human intrusion, rather than establishing fixed time frames it is more useful to identify distinct stages in the long term evolution of the disposal system and the decay of its radionuclide inventory and to relate these to the consequences of human intrusion.

There is a need to establish clearly the distinctions and relationships between human intrusion, retrievability, safeguards and institutional control, for example, retrieval can be seen as a legal action and is not equivalent to human intrusion which is unauthorised and sometimes illegal.

International standards

In relation to a position for inclusion in international standards, the presentations and discussion showed that there is agreement on several general principles in relation to human intrusion and that there may be the basis for a common approach for treating human intrusion in safety assessments, but that more needs to be done to elaborate such a common approach. Whether this can be developed within the time frame of the current standards development programme remains to be seen.

2.6. Reversibility and retrievability

The Chairman of this session (P. Raimbault) noted that reversibility and retrievability are now being considered by several countries in association with their geological disposal programmes.

Reversibility is the possibility of reversing one or a series of steps in repository planning and it is coherent with the stepwise approach.

Retrievability is the possibility of reversing the action of waste emplacement.

Reasons to provide retrievability

There are several possible reasons for wishing to provide for retrievability of waste. The main reason is to help to build public confidence, but the waste (especially if it is spent fuel declared as waste) may become a valuable resource in the future. Retrievability may be considered to be consistent with the ethical consideration that future generations should be free to make their choice.

Implications for safety

With respect to its implications for safety, making provision for retrievability is likely to imply that repositories will be left open for much longer periods than if it were not provided for. There may then be a higher probability of human intrusion into the repository and of possible abandonment without closure. On the other hand the need to provide for safety during the open retrievable period may imply that more robust and durable packages will be developed and it can be argued that the additional information obtained during the extra period during which the repository is open could improve the safety case. The retrievability option may be difficult to achieve for disposal in salt where the gradual and automatic closure of vaults by creep is an important feature of the safety case.

International standards

With respect to the treatment of retrievability in regulations and international standards, it is noted that some legislations or regulations already mention that the retrievability option should be considered in disposal concepts. However, it may be too early to produce precise safety requirements with respect to this option. In principle, any measure to enhance retrievability should not compromise operational or long term safety. (It has to be recognized that it may be difficult to judge, in advance, a potential compromise to operational safety and that measures to make the waste more accessible are almost bound to lead to increased worker exposures.) The recent NEA booklet on reversibility/retrievability provides a good discussion of the subject.

2.7. Monitoring and institutional control

The Chairman of this session (D. Metcalfe) began by summarizing the main technical areas of agreement:

Key points of technical agreement

Geological repositories are designed and implemented to avoid the need for long term monitoring and institutional control. They will not be closed until there is a high degree of assurance of long term safety. Nevertheless, monitoring and institutional controls can contribute to the achievement of long term safety by helping to prevent human intrusion and, in the event of its occurrence, by mitigating the consequences. A fundamental condition in this context is that monitoring and institutional control measures including provisions for safeguards surveillance can only be permitted if they do not detract from the long term safety of the repository.

Reasons for post closure action

(1) institutional controls (local land use controls, records and possibly markers) will help to prevent and mitigate the impacts of human intrusion;

(2) institutional controls (surveillance, inspection and monitoring) will be needed to satisfy the requirements of international nuclear safeguards agreements;

(3) post closure monitoring of the facility and its environs is likely to be required to provide reassurance to the local population that the facility is performing as planned;

(4) post closure monitoring may be carried out for the purpose of accumulating an environmental data base on the repository site and its surroundings that may be of use to future decision makers;

(5) post closure monitoring may be used to strengthen understanding of some aspects of system behaviour used in developing the safety case for the repository and to allow further testing of models predicting those aspects.

Views of local communities

Over the years, it has become increasingly evident that local communities consider post closure monitoring and institutional controls to be evidence of good stewardship on the part of the implementor or government. Contrary to past thinking, when it had been thought by many that long term monitoring would be seen by the public as evidence of a lack of confidence in long term safety. The public expects such measures and does not consider them to be an undue burden on future generations. They can be seen and presented in various positive lights, for example, as part of a national environmental data base.

Responsibility of each generation

It has to be recognized that the current generation has a limited ability to ensure that disposal following the current plans is implemented and that each generation will make its own plans with respect to the use and closure of the repository. Each generation has the responsibility to pass down to the next generation the information, skills and knowledge needed to make informed decisions. Further consideration needs to be given to mechanisms for achieving the transfer or passing down of information between generations.

International standards

The presentations and discussion showed that although there are difficulties in developing a fully satisfactory rationale for managing affairs in the post closure period and that the position continues to change, nevertheless, there are some clear positions to include in the standards (as listed in these conclusions). Further consideration needs to be given to the implications of the requirements for nuclear safeguards and to the means for preserving long term institutional control. In relation to monitoring, the recent IAEA-TECDOC-1208 (Monitoring of Geological Repositories for High Level Radioactive Waste, 2001) provides a good discussion of the subject.

ANNEX
PRESENTATIONS AT THE SPECIALISTS MEETING

A.1. INTRODUCTORY SESSION

ICRP GUIDANCE ON RADIOACTIVE WASTE DISPOSAL

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Abstract. The International Commission on Radiological Protection (ICRP) issued recommendations for a system of radiological protection in 1991 as the 1990 Recommendations. Guidance on the application of these recommendations in the general area of waste disposal was issued in 1997 as Publication 77 and guidance specific to disposal of solid long-lived radioactive waste was issued as Publication 81. This paper summarises ICRP guidance in radiological protection requirements for waste disposal concentrating on the ones of relevance to the geological disposal of solid radioactive waste. Suggestions are made for areas where further work is required to apply the ICRP guidance.

1. INTRODUCTION

The International Commission on Radiological Protection (ICRP) issued its latest general recommendations for protection against ionising radiation in 1991 as the '1990 Recommendations' [1]. In 1997, ICRP elaborated on these general recommendations in the context of radioactive waste issuing its 'Radiological protection policy for the disposal of radioactive waste' [2]. This policy addressed all types of radioactive waste, short-lived, long-lived, solids, liquids and gases. It was followed, in 2000, by Publication 81 which developed recommendations specifically for the disposal of long-lived solid radioactive waste [3].

This paper describes the key aspects of these ICRP recommendations focussing particularly on the ones of relevance to geological disposal of long-lived waste. Emphasis is placed on the criteria and principles that would apply to new purpose built facilities

2. THE 1990 RECOMMENDATIONS

These provide ICRP's fundamental recommendations for protection of people from ionising radiation. They are intended for general application, covering the full range of circumstances of exposure, only those of relevance to radioactive waste disposal are addressed here.

The 1990 Recommendations distinguish between two types of exposure situations: practices and interventions. Practices are deliberate human activities which, as a by-product, result in increased exposure of individuals or populations and, in principle, can be designed and operated to meet radiological protection requirements that are specified in advance. Interventions are human activities intended to decrease overall exposure and apply in situations in which the source of exposure is already present when decisions on protective actions are to be taken. The basic principles of radiological protection are used to establish the levels of control of exposure in both practices and interventions but are applied in different ways.

This paper is largely concerned with the principles of protection for practices, which are in summary:

Justification	– overall a practice should do more good than harm
Optimisation	– the margin of good over harm should be maximised
Dose and risk limitation	– an adequate standard of protection should be provided for the most exposed individuals

Additionally, in applying the principle of protection for optimisation a constraint should be placed on the maximum dose or risk incurred by individuals. This constraint should correspond to a value less than the dose limit for members of the public. ICRP, however, in Publication 60 did not recommend a value for the constraint. The constraint is applied prospectively; options implying doses or risks higher than the constraint should be rejected.

The basic tenet underpinning these principles is the linear-no threshold hypothesis.

In many situations it is virtually certain that exposures will occur and their magnitude will be predictable, albeit with some degree of uncertainty. These are referred to as 'normal exposures'. Conversely, there may be circumstances where exposures do not occur as planned or where there is a potential for exposure, but no certainty that it will occur. ICRP calls such exposures 'potential exposures' recommending that they should be taken into account when considering the need for protective actions.

Exposures arising in the future following the disposal of solid radioactive waste will depend on events and processes that have probabilities associated with them and, as such, would be potential exposures.

3. PUBLICATION 77: RADIOLOGICAL PROTECTION POLICY FOR THE DISPOSAL OF RADIOACTIVE WASTE

This publication extended and developed the 1990 Recommendations for specific application to the disposal of all types of radioactive waste, including effluents as well as solids. It was produced in response to a perceived need to clarify the application of the 1990 Recommendations in this area. Particular issues included application of the justification and optimisation principles to waste disposal in circumstances where doses to individuals and populations may be projected to be received over extended periods of time into the future and with increasing associated uncertainties. The use of the quantity collective dose in this context was a particular issue as was the type of criteria that should be applied in establishing whether individuals in future populations were being afforded an adequate degree of protection.

In general, ICRP considers that the main radiological protection issue in waste disposal is public exposure. This should be controlled through the process of constrained optimisation. A value for the dose constraint for members of the public of no more than about 0.3 mSv in a year is suggested to be appropriate. Use of this process, it is suggested by ICRP, would obviate the direct use of dose limits for members of the public and as such, in this context, they should progressively fall out of use.

ICRP clarified application of the principles of justification and optimisation. In the case of justification, as waste management and disposal operations are an integral part of the practice generating the waste, it is wrong to regard them as a free standing practice that needs its own justification. It follows that consideration should be given to such operations in the process of deciding whether the practice generating the waste is justified or not. In the context of optimisation, ICRP considered that this principle had become too closely linked to quantitative techniques such as differential cost benefit analysis. Whereas, ICRP's policy is more judgmental being summarised, in essence, by the following statement taken from Publication 55 [4], "Have I done all that I reasonably can to reduce these radiation doses?"

The use of the quantity collective dose is linked to the application of the principles of justification and optimisation being a broad indicator of detriment to the exposed population. Collective dose is an aggregated quantity and problems arise by use of collective doses from very small doses to large populations and from doses occurring over very long times. In the case of very small doses, ICRP does not recommend that the component of collective dose due to small individual doses should be ignored solely because the individual doses are small. Where possible, however, it would provide more information for decision-making purposes to separately identify that component of the total collective dose which is delivered at very small individual doses; decision-makers could, for example, place lower weight in the decision-making process on this component versus the components of collective doses that are delivered at larger individual doses. The weight attached to collective doses delivered over long time periods into the future was also addressed by ICRP. Uncertainty is the main problem. Both the magnitude of the individual doses and the size of the exposed population become increasingly uncertain as the time increases. There is also the issue that current judgements about the relationship between dose and health effects (or detriment) may not be valid for future generations. In the light of such arguments, ICRP suggests that forecasts of collective doses over periods longer than several thousand years and forecasts of health detriment over periods longer than several hundreds of years 'should be examined critically'.

Overall, ICRP argues that when collective doses are being considered in decisions on waste management options, it is more informative to present the doses in blocks of individual dose and time intervals.

On the question of judging the significance of possible harm to future generations from decisions taken today, ICRP investigated a number of quantities including individual doses and risks, and collective doses. The conclusion was that the appropriate quantities to use are the calculated annual individual doses and risks in the relevant critical groups.

ICRP, in Publication 77, makes important observations about the practical implications of considering potential exposure in waste management decisions. The possibility of potential exposure distinguishes between the two waste disposal strategies 'Dilute and Disperse' and 'Concentrate and Retain'. Direct discharge of radioactive waste to the environment, the dilute and disperse option gives rise to more or less immediate critical group doses whose magnitude can be assessed reasonably reliably. Potential exposure is usually not an issue. The retention strategy usually eliminates or significantly reduces public exposures at least in the short to medium term. It can appear, therefore, the more protective of the two strategies. Consideration of potential exposure may influence this view as events with probabilities associated with them could lead to significant exposures. For example, there is the possibility, which is discussed in more detail later, of inadvertent human intrusion into the waste. ICRP recommends that waste disposal policies should take account of potential and normal exposures. Furthermore, the dispersion of radioactive waste should not automatically be regarded as a less suitable option than retention. For solid waste, however, disposal in a more or less undiluted form is often the only viable option.

4. PUBLICATION 81: RADIOLOGICAL PROTECTION RECOMMENDATIONS AS APPLIED TO THE DISPOSAL OF LONG-LIVED SOLID RADIOACTIVE WASTE

This further develops the 1990 Recommendations [1] and the policy for the disposal of radioactive waste [2] specifically in the context of the disposal of solid long-lived radioactive waste. The disposal options considered are surface or near surface disposal and geological disposal. All types of solid long-lived radioactive waste are covered in Publication 81 including solid high level waste and large volume low level waste. Issues addressed include the radiological protection criteria that should be applied in establishing whether future populations are being adequately protected, and guidance on how to apply the optimisation principle in waste disposal. Particular consideration is given to evaluating the significance of exposures arising as a result of direct inadvertent human intrusion into a waste disposal facility. This publication also extends and develops proposals in an earlier ICRP publication, ICRP 46 [5], which also addressed disposal of solid radioactive waste.

ICRP considers, in Publication 81, that the main protection issues concern exposure of members of the public that may or may not occur in the future. Furthermore, individuals and populations in the future should be afforded at least the same level of protection from the action of disposing of radioactive waste today as is the current generation. This implies the use of the current dose and risk criteria derived from considering associated health detriment. Therefore, in principle, protection of future generations should be achieved by applying these dose and risk criteria to the estimated future doses and risks in appropriately defined critical groups. These criteria are dose limits and dose and risk constraints. However, as it is impossible to know in advance what the total dose to future individuals will be from practices, the appropriate criteria are the constraints: constrained optimisation is the central approach to evaluating the radiological acceptability of a waste disposal system.

Knowledge of the disposal facility may be lost in the future and, consequently, it cannot be assumed that any mitigation measures would be carried out to reduce doses should these reach unacceptable levels. An effective disposal facility will, however, retain the waste during the period of greatest potential hazard with only residual radionuclides entering man's environment in the distant future. In this future time period, two broad categories of exposure situation are distinguished: natural processes and human intrusion. The term human intrusion covers inadvertent human actions affecting repository integrity and potentially having radiological consequences. It is more likely to occur after knowledge of the repository has been lost, *ie*, in the far future. The term, natural processes, includes all the processes that lead to the exposure of individuals other than human intrusion. Application of radiological protection criteria to these two categories of exposure situations is different.

Natural Processes. These processes include the foreseen gradual degradation of the repository together with other, less likely, natural processes that may disrupt the performance of the repository. Therefore the objective of protecting the public in such circumstances would have to consider both the probability of occurrence and the magnitude of the corresponding exposures. This can be achieved by either aggregating the probabilities and corresponding doses (or rather the risk equivalent of the dose) in an overall evaluation of risk or by separate consideration of the dose and associated likelihood of occurrence of the exposure in a disaggregated approach. The key criterion is the dose constraint for members of the public. An upper numerical value of 0.3 mSv y^{-1} has been recommended by ICRP (see section 3); this corresponds to a risk constraint of the order of 10^{-5} per year. In the aggregated approach, the total risk to a representative critical group is compared with the risk constraint. This is conceptually satisfying but requires a comprehensive evaluation of all relevant exposure situations and their associated probabilities of occurrence; a process that can be difficult to achieve in a transparent and convincing manner. In the disaggregated approach, likely or representative release scenarios are identified and the calculated doses from these scenarios are compared with the dose constraint. The radiological significance of other, less likely, scenarios can be evaluated from a separate consideration of the resultant doses and their probabilities of occurrence. ICRP considers that although a similar level of protection can be achieved by these approaches, more information can be obtained for decision-making purposes from the disaggregated approach.

All of these approaches require assessments of doses or risks to critical groups. Due to the long timescales under consideration, the habits and characteristics of the critical group as well as those of the environment in which it is located can only be assumed. Thus, the critical group is hypothetical. It should be chosen on the basis of reasonably plausible assumptions taking account of current lifestyles and site or region specific information. In many cases different exposure scenarios with their associated critical groups will have different probabilities of occurrence and therefore the highest dose may not correspond to the highest risk. Because of this, ICRP suggests that it is important to clearly present the different exposure scenarios and their associated probabilities of occurrence in the decision-making process.

The long timescales under consideration impact on the importance attached to the assessed doses and risks. These estimates, according to ICRP, should not be regarded as measures of health detriment beyond times of around several hundreds of years into the future. In the case of longer time periods, they represent indicators of the protection afforded by the disposal system.

Human Intrusion. The possibility of elevated exposures from human intrusion is an inescapable consequence of the decision to concentrate waste in a discrete disposal facility. An intrusion event could result in radioactive material being brought to the surface resulting in the exposure of nearby populations to significant radiation doses. How should such events be taken into account in evaluating the radiological acceptability as a waste disposal option?

In principle, a risk based approach, considering both the probability and consequences of human intrusion, could be used to evaluate the radiological significance of human intrusion. ICRP, however, cautions against this approach as there is no scientific basis for predicting the nature or probability of the corresponding future human actions. Instead ICRP suggests that it is the radiological consequences of intrusion that should be considered. It is emphasised, however, that these consequences should not be compared with the dose constraint of 0.3 mSv per year for members of the public; this constraint applies during the process of optimisation of protection and, by definition, intrusion will have by-passed all of the barriers which were considered during the optimisation process. So, what should the consequences be compared with? ICRP considers that in circumstances where human intrusion could lead to doses to those living around the site sufficiently high that intervention on current criteria would almost always be justified, reasonable efforts should be made to reduce the probability of intrusion or to limit its consequences. General criteria for intervention in the case of long-term exposures have been established in ICRP 82 [6] and ICRP suggests that these could be used to evaluate the significance of human intrusion. Briefly, these criteria are that an existing annual dose of 10 mSv may be used as a generic reference level below which intervention is not likely to be justifiable, whereas at 100 mSv per year and above, intervention is considered to be almost always justifiable. The term existing annual dose refers to the sum of the existing and persisting annual doses to individuals at a given location. The exposure that may occur from a repository is one component of this. The doses should be assessed using plausible stylised exposure scenarios representing human intrusion events.

This guidance from ICRP addresses the exposures to individuals in local population groups but there is the issue of the exposure of the intruder. This is not directly addressed by ICRP.

Technical and Managerial Aspects. ICRP considers that the technical and managerial principles developed generally for potential exposure situations in ICRP 64 [7] should be applied during disposal system development to enhance confidence in radiation safety. These principles should be applied to disposal systems in a manner consistent with the inherent level of hazard of the waste as well as with the level of residual uncertainty identified in the assessment.

One of the key principles is the concept of defence in depth which provides for successive passive safety measures which enhance confidence that the disposal system is robust and has an adequate safety margin. In addition to the technical principles, an essential managerial principle for all individuals and organisations involved in the repository development process is to establish a consistent and pervading approach to safety which governs their actions.

Compliance with Radiological Criteria. This is a difficult topic because of the inherent uncertainties in our estimates of radiological impacts on future generations. For this reason, demonstration that radiological protection criteria will be met in the future is not as simple as a straightforward comparison of estimated doses and risks with the appropriate constraints. Judgement may be required. ICRP points out that the dose and risk constraints should be considered as reference values for time periods beyond several hundreds of years into the future, and additional arguments should be brought to bear when judging compliance. A decision on the acceptability of a disposal system should be based on reasonable assurance rather than on an absolute demonstration of compliance with numerical criteria.

Overall, ICRP's view on compliance is that provided that the appropriate constraint for natural processes has been appropriately satisfied, that reasonable measures have been taken to reduce the probability of human intrusion and that sound engineering, technical and managerial principles have been followed, the radiological protection requirements can be considered to be satisfied.

5. CONCLUSIONS

From its general Recommendations issued in Publication 60, ICRP has developed a framework for evaluating the radiological acceptability of options for the disposal of all types of radioactive waste. The central process is one of constrained optimisation. To be used, this framework requires elaboration in a number of areas by the appropriate national or international organisations. These areas include:

— Characterisation of critical groups and the associated biospheres for use in estimating doses or risks from *natural processes.*
— Development of stylised human intrusion scenarios for evaluating the significance of *human intrusion.*

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A.2. COMMON FRAMEWORK FOR RADIOACTIVE WASTE DISPOSAL

A COMMON FRAMEWORK FOR THE SAFE DISPOSAL OF RADIOACTIVE WASTE

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Abstract. Various industrial, research and medical activities give rise to waste that contain or are contaminated with radioactive material. In view of the potential radiological hazards associated with such waste they have to be managed and disposed of in such a way as to ensure that such potential hazards are adequately managed and controlled in compliance with the safety principles and criteria. Over the past few decades experience in radioactive waste management has led to the development of various options for radioactive waste management and has also led to the development of principles which the various waste management options should satisfy in order to achieve an acceptable level of safety. International consensus has evolved in respect of the principles. However, complete consensus in respect of demonstrating compliance with the requirements for managing and disposing of the whole range of waste types is still developing. This paper identifies the various waste types that have to be managed, the prevailing safety principles and the disposal options available. It discusses the development of a common framework which would enable demonstration that a particular disposal option would meet the safety principles and requirements for the disposal of a particular waste type.

1. INTRODUCTION

Various industrial, research and medical activities give rise to waste that contain or are contaminated with radioactive material. In view of the potential radiation hazards associated with such waste they have to be managed and disposed of in such a way as to ensure that such potential hazards are adequately controlled. The different associated processes including disposal are generally referred to as radioactive waste management. Over the past few decades experience accrued in radioactive waste management has led to the development of various options for such management and has also led to the development of principles which the various waste management options should satisfy in order to achieve a good level of safety. International consensus has evolved in respect of the principles. However, complete consensus over the actual requirements for managing and disposing of the whole range of waste types is still developing.

One of the main generators of radioactive waste is the nuclear fuel cycle, this is the sequence of steps in the nuclear energy generation process starting with the mining of ores rich in naturally radioactive uranium and thorium. These elements have radioactive half-lives comparable to the age of the earth and make up on average some ten parts per million of the earth’s crust. As with many other elements there are localized concentrations in the earth’s crust and mineralized deposits comprising up to twenty percent uranium have been found to exist. Concentrations ranging from around a few hundred parts per million upwards have been commercially exploited for the extraction of uranium. Mined ores go through mineral extraction processes which include crushing, milling and metallurgical extraction. The concentrates are then usually transported in the form of uranium oxides to refining and conversion facilities where the purified uranium is converted to uranium hexafluoride for enrichment. The enriched uranium is subsequently converted, usually to uranium dioxide, which is then fabricated into nuclear fuel, the exact fuel design depending on the reactor type in which the fuel will be used. Spent nuclear fuel is removed from reactors and placed in storage for decay pending subsequent disposal. The spent nuclear fuel contains the highest concentrations of radioactive materials and repositories for its final disposal are still in the process of development. Whilst the concentrations of radioactive materials are greatest in the spent fuel, each step in the nuclear fuel cycle gives rise to radioactive waste in one form or another. These vary widely in radionuclide type and content, and in physical and chemical form and in quantity.

Nuclear energy may be exploited in nuclear power stations to produce electricity, it may also be exploited in research and radioisotope production reactors. In the latter, the nuclear fission process is used to produce radiation beams for purposes of research and development or for the production of radioactive material which are used for a broad range of applications. Radionuclides can also be produced in accelerators and together with reactor produced radionuclides are referred to as artificial radionuclides. Such radionuclides can be fabricated into sealed radiation sources or used in an unsealed form and both find widespread applications in industry, medicine and research. Again the manufacture and use of these sources gives rise to the generation of radioactive waste and again the characteristics of the waste can vary widely depending on the particular radionuclides involved, their associated radioactive half lives and the activity of the sources. The nature of the use for which the source was employed will also greatly influence the physical and chemical characteristics of the waste and the amounts of material involved.

In addition to the various activities discussed above where materials are exploited for their fissionable properties, such as uranium, or their radioactive properties in the case of radiation sources, other activities involving materials containing elevated levels of naturally occurring radionuclides generate radioactive waste. As with the former activities, the latter also give rise to a broad range of waste types with varying radioactive, physical and chemical characteristics.

The one aspect in common with all radioactive waste, regardless of their type or origin is their radioactive nature and their potential to give rise to radiological hazards, albeit of widely varying magnitude. Nevertheless, they all need to be managed as radioactive waste, with appropriate levels of control, if their safety is to be assured.

The safety principles developed for radioactive waste management and the various technologies developed to ensure these principles are met have largely been based on consideration of waste arising from the nuclear fuel cycle and largely on the waste arising from the back-end of the cycle i.e. operational waste and spent nuclear fuel arising from nuclear power stations. More recently a number of anomalies have become more evident in application of the principles to the broader range of waste types and the circumstances of their generation and management, particularly in respect of waste arising from the front end of the nuclear fuel cycle, those associated with naturally occurring radioactive materials and in the disposal of spent or disused radiation sources. In order to address these anomalies and to ensure that all radioactive waste are managed in an acceptably safe manner, it has been suggested that a common framework should be established to provide for the safe management of all radioactive waste types. The common framework should identify all radioactive waste types, the agreed safety principles, the issues to be considered in their management and how the safety of a particular disposal option can be can be reasonably assured.

2. RADIOACTIVE WASTE TYPES AND CHARACTERISTICS

The broad range of waste types that are generated can be managed in a variety of ways and it is convenient to classify waste for which particular management arrangements are appropriate. The determination of what management arrangements are appropriate requires consideration of a number of factors including the radiological, physical and chemical properties of the waste, the amounts and timeframes of waste generation, the processing capabilities available and the disposal options available. Nevertheless, it is possible to identify a number of generic disposal options that are or could be made available and a few broad categories of waste types that with appropriate processing would logically be compatible with a particular disposal option. The waste types and their characteristics are tabulated below.

Waste Type	Origins	Characteristics
Residues and waste from mining and processing of radioactive ores	• Uranium mining • Thorium mining • Minerals sands mining • Minerals with elevated concentrations of naturally occurring radionuclides – phosphates, copper, fluorspar, coal etc.	Large volumes Low specific activities Very long radioactive half lives Radon gas generation Contaminated scales Contaminated equipment – metal, timber, plastic etc. Adventitious concentrates
Waste containing higher concentrations of longer lived alpha emitting waste	• U refining, conversion, enrichment and fuel fabrication • Minerals processing – rare earth extraction, oil refining, fertilizer production, water treatment etc.	Chemical residues – sludges, resins, carbon Filters Process tailings Scales Contaminated equipment – metal, plastic Contaminated bricks and concrete Long half lives Adventitious concentrates
Spent fabricated radiation sources with short radioactive half live < 1 year	Industry, medicine and research	Small size Short half lives High specific activity Sealed /unsealed
Spent fabricated radiation sources with radioactive half lives > 1 year < 30 years	Industry, medicine and research	Small size Medium half lives High specific activity Sealed
Spent fabricated radiation sources with longer radioactive half lives > 30 years	Industry, medicine and research	Small size Long half lives High specific activity Sealed
Operational waste from nuclear installations and other industrial/medical/research activities containing primarily radionuclides with radioactive half lives shorter than 30 years and with activity concentrations not requiring significant radiation shielding.	• Nuclear power stations • Research reactors • Nuclear fuel cycle facilities • Research laboratories • Hospitals	Resins Filters Chemical concentrates and sludges Contaminated consumable – clothing, plastic etc. Contaminated equipment Decommissioning waste Fission and activation products Medium to low specific activities
Operational waste from nuclear installations and other industrial/medical/research activities containing primarily radionuclides with radioactive half lives shorter than 30 years and with activity concentrations requiring radiation shielding.	• Nuclear power stations • Research reactors • Nuclear fuel cycle facilities • Research laboratories • Hospitals	Filters Chemical concentrates and sludges Contaminated consumable – clothing, plastic etc. Contaminated equipment Decommissioning waste Fission and activation products Medium to low specific activities Biological waste
Spent nuclear fuel and high level waste from reprocessing of spent nuclear fuel	• Nuclear power stations • Research reactors • Re-processing facilities	High specific activity Fission products, actinides

3. SAFETY PRINCIPLES

A number of bodies have played a role at an international level in the development of safety principles for radioactive waste safety. Prime amongst these bodies are the International Commission on Radiological Protection and the International Atomic Energy Agency. The former is an independent international commission which makes recommendations based on prevailing scientific knowledge and the IAEA is a specialist UN Agency within whose mandate it is to establish international consensus on safety standards for radioactive waste safety together with radiation and nuclear safety. To assist the IAEA in these functions it has established a Commission on Safety Standards (CSS) with supporting specialist committees e.g. Waste Safety Standards Committee (WASSC), and an International Nuclear Safety Advisory Group.

3.1. IAEA Principles

In 1995, the Agency issued a Safety Fundamentals publication entitled "Principles of Radioactive Waste Management". These principles are referred to in the following sections and are derived from a stated objective of radioactive waste management, namely:

"*to deal with radioactive waste in a manner that protects human health and the environment now and in the future without imposing undue burdens on future generations*".

The principles that are particularly relevant to the discussion in this document are as follows:

Principle 1: *"Radioactive waste shall be managed in such a way as to secure an acceptable level of protection for human health."*

This has generally been interpreted simply to mean that, in general terms, radioactive waste is to be managed according to the normal system of radiation protection for practices. This is typically qualified by noting that waste management on its own does not require justification, but that the justification of the practice or intervention that gives rise to the waste should include consideration of the management of the waste.

Principle 2: *"Radioactive waste shall be managed in such a way as to secure an acceptable level of protection of the environment."*

The extent to which this has been interpreted as distinct from Principle 1 varies. Of interest to this paper is the statement, in elaborating upon the text of the principle itself, that: "The preferred approach to radioactive waste management is concentration and containment of radionuclides rather than dilution and dispersion in the environment. The most recent ICRP recommendations are less supportive of such a preference.

Principle 4: *"Radioactive waste shall be managed in such a way that predicted impacts on the health of future generations will not be greater than relevant levels of impact that are acceptable today."*

Variants of this principle are widely used in national and international recommendations, but this formulation is more explicit than most (other versions refer, for example, to a level of protection at least equivalent to that provided today). Regardless of the wording, this principle has most commonly been interpreted to mean that the same numerical criteria (limits or constraints) should be applied for all times in the future. Firstly, it should be noted that even this quite rigid interpretation leaves a very large degree of uncertainty as to the level of assurance needed. We have a relatively good idea of the doses that people actually receive at present, but the doses to individuals in future generations are so dependent upon aspects of their environment and behaviour which have such large associated uncertainties as to be not "predictable" in any meaningful sense. Furthermore, there is no certainty that a given dose will have the same impact on health in the future. Typically, however, the practical interpretation has been to estimate doses to hypothetical humans with all the characteristics existing today transferred into the future. Although this is a convenient method for obtaining a quantitative measure of the "predicted impact", it is not at all clear what the results really mean. Secondly, the measures used to represent the "levels of impact that are acceptable today", such as the dose constraint applied to discharges, do not fully represent the level of protection that exists today. The present generation may experience radiological impacts not only from the planned operation of current practices, but also from the residues of past practices and from accidents that might occur in current practices (although there are mechanisms in place to intervene if necessary to mitigate the effects of accidents). It is not a straightforward task to determine what situation in the future, viewed from the perspective of the present day, would be equivalent to this. Clearly, the difficulties in applying this principle increase as the time periods of concern increase, and so are particularly pertinent to the management of very long-lived waste.

Principle 5: *"Radioactive waste shall be managed in such a way that will not impose undue burdens on future generations."*

This principle is intended to relate to non-radiological "burdens". It has been generally understood to mean that the present generation should do as much as it can to provide for the safe long term management of the waste it generates, leaving as little as possible for future generations to do. One example quoted in the safety fundamentals is that "*the management of radioactive waste should, to the extent possible, not rely on long term institutional control as a necessary safety feature*". In practice, however, consideration of long term institutional control has tended to focus more on the reliance that can be placed on it than on whether it represents an undue burden. This raises fundamental questions in relation to the disposal of very long-lived waste in places where, in the absence of institutional control, they could relatively easily reach the human environment.

There are four remaining principles that are not relevant to the present discussion. For completeness, these are listed below:

Principle 6: *"radioactive waste shall be managed within an appropriate national legal framework including clear allocation of responsibilities and provision for independent regulatory functions."*

Principle 7: *"generation of radioactive waste shall be kept to the minimum practicable."*

Principle 8: *"interdependencies among all steps in radioactive waste generation and management shall be appropriately taken into account."*

Principle 9: *"the safety of facilities for radioactive waste management shall be appropriately assured during their lifetime.*

3.2. INSAG principles

The IAEA has established an International Nuclear Safety Advisory Group (INSAG), formed by well known experts, which has the function of providing advice to the agency's director general. The eleventh INSAG report draws together the three safety fundamentals publications in the safety series dealing respectively with the safety of nuclear installations, radioactive waste and radiation sources.

Due to its brevity the report cannot provide much detail, but, regarding the disposal of radioactive waste, none of the relevant principles described in section 3.1.1 is in any way contradicted. In general terms the point is stressed that disposal options need to be commensurate to the activity level and the longevity of the radiological hazard. In particular paragraph 91 says:

“*for waste disposal the objective is to achieve an ultimately passive solution with, as far as possible, no long term requirements for intervention by humans or for continuing institutional control. Disposal thus seeks to isolate the waste from the environment for sufficiently long periods of time so that the risks to humans from such disposal, including any risk from inadvertent human intrusion, would be very small.*”

This can be interpreted as supporting at once several of the principles quoted above. Then the report goes on to stress the importance of defence in depth and quality control.

3.3. Recent Developments in ICRP Recommendations

Since publication of the IAEA safety fundamentals, ICRP has issued three publications relevant to waste disposal that extend and expand on the system of protection recommended in ICRP Publication 60. These publications are:

(1) Radiological Protection Policy for the Disposal of Radioactive Waste, ICRP Publication 77

(2) Radiation Protection Recommendations as Applied to the Disposal of Long-lived Solid Radioactive Waste, ICRP Publication 81

(3) Protection of the Public in Situations of Prolonged Radiation Exposure, ICRP Publication 82.

The important aspects of these publications are discussed below followed by a summary of their implications for the management of radioactive waste in general.

3.3.1. Summary of ICRP Publications

In publication 77, ICRP deals with specific policy issues that arise from application of its system of protection to the disposal of radioactive waste whether in the form of gaseous or liquid discharges or as solids. ICRP reaffirmed that the principles of protection for practices would normally be applied to the controlled disposal of radioactive waste and that the main issues concern exposure of the public. In the context of waste management strategies, ICRP considers that ‘the dispersal of radioactive waste should not be automatically regarded as less suitable than retention’.

The main implications of this publication concern optimization of protection and use of the quantity collective dose. In publication 77, the process of optimization of protection is considered to have a strong subjective element being ‘more subtle and judgmental than is implied by differential cost–benefit analysis’. Constraints should be applied during the optimization of protection and ICRP recommends an upper value for the dose constraint for members of the public from waste management operations of 0.3 mSv/a. Constrained optimization is considered to be the means for controlling public exposure from waste management operations.

In connection with the quantity collective dose, ICRP considers that estimates for time periods longer than several thousands of years into the future should be viewed with caution. One purpose for calculating collective dose is to estimate possible numbers of health effects (referred to by ICRP as ‘health detriment’). Taking account of the fact that the currently understood relationship between dose and risk might not be valid for future generations, ICRP takes the view that forecasts of health detriment over periods longer than several hundreds of years into the future should be viewed critically. Overall, in the context of protection of future generations from decisions taken today, ICRP suggests that estimates of doses or risks to individuals in future critical groups will provide a basis for deciding on the appropriate level of protection.

ICRP's policy, as enunciated in Publication 77, was developed in the context of disposal of long lived solid radioactive waste in Publication 81, taking account of ICRP's previous recommendations in this area. Publication 81 also applies some of the principles developed in Publication 82 which is described later.

Publication 81 states that the principal objective of disposal of solid radioactive waste is the protection of current and future generations from the radiological consequences of waste produced by the current generation. Furthermore, ICRP acknowledges the basic ethical principle that individuals and populations in the future should be afforded at least the same level of protection from actions taken today as is the current generation. This parallels at least one principle in the safety fundamentals. In evaluating the level of protection being afforded future generations, ICRP concludes that two broad categories of exposure situations must be considered: natural processes and inadvertent human intrusion.

In the first case, estimated doses and risks should be compared with a dose constraint of no more than about 0.3 mSv/a or its risk equivalent. In the case of human intrusion, ICRP recommends that the consequences of intrusion are assessed in terms of doses to exposed individuals using one or more stylized human intrusion scenarios. It is pointed out that it is not appropriate to compare these assessed doses with the dose or risk constraints for natural processes because by definition 'intrusion will have by-passed all of the barriers which were considered in the optimization of protection'. Instead, ICRP recommends that they are compared with current criteria used to establish whether intervention is necessary; if human intrusion could result in doses 'to those living around the site sufficiently high that intervention on current criteria would almost always be justified, reasonable efforts should be made during the design and development of the repository to reduce the probability of intrusion or to limit its consequences'. The criteria for deciding whether intervention would be justified are taken from Publication 82.

Publication 81 addresses the issue of institutional controls. Whilst acknowledging that *'it cannot be assumed that future generations will have knowledge of disposals undertaken by the current generation'* ICRP points out *'that institutional controls maintained over a disposal facility after closure may enhance confidence in the safety of the disposal facility particularly by reducing the likelihood of intrusion'*. Of particular relevance to the present discussion, ICRP goes on to state that *'there is no reason why these controls may not continue for extended periods of time and, therefore, may make a significant contribution to the overall radiological safety of shallow facilities in particular. Furthermore, for surface or near surface disposal of uranium mill tailings, these controls may be relied upon for long periods of time in situations where, if the controls fail, consequences will be generally lower than those associated with other long-lived radioactive waste.'*

Developing on the points made in Publication 77, it is considered that estimates of doses and risks can only be regarded as measures of health detriment for times up to around several hundreds of years into the future. For longer time scales they only represent indicators of the protection afforded by the system. This point is taken up by ICRP in considerations of compliance with numerical criteria such as dose or risk constraints where it is stated that *'as the time frame increases, some allowance should be made for assessed dose or risk exceeding* *the dose or risk constraint'*. again, the quantity collective dose is regarded as having little or no role to play.

Constrained optimization is considered to be the key to establishing the radiological acceptability of a waste disposal system. Publication 81 further extends the judgmental approach to optimization of protection noted in Publication 77. Publication 81, in the context of disposal of long lived solid radioactive waste, states that *'optimisation of protection is a judgmental process with social and economic factors being taken into account….the goal is to ensure that reasonable measures have been taken to reduce future doses to the extent that required resources are in line with these reductions'*. ICRP makes further use of the idea of reasonableness in its concluding recommendations on the evaluation of the radiological acceptability of a disposal option by stating: *'provided reasonable measures have been taken both to satisfy the constraint from natural processes and to reduce the probability or consequences of inadvertent human intrusion, and technical and managerial principles have been followed, then radiological protection requirements can be considered to have been complied with'*. This statement is only relevant if calculated exposures remain below some standards, for example intervention levels.

Publication 82 deals with the application of ICRP's system of protection to radiation exposure from natural sources and long lived residues. It derives criteria from, amongst other things, consideration of natural background levels, for deciding whether intervention may be justified in such situations: in circumstances where exposures are above 100 mSv/a, intervention is considered to be almost always justified, conversely in situations where exposures are below 10 mSv/a intervention is probably not justified. (these criteria are also used in Publication 81 to assess the significance of inadvertent human intrusion into a waste repository — see above). The report discusses some issues of relevance to the management of a particular category of waste, *i.e.* waste with elevated levels of naturally occurring radionuclides. It emphasizes that residues remaining from past activities should be treated within an intervention framework. For proposed new facilities generating such waste, application of optimization of protection within dose and risk constraints is recommended with the proviso that in some cases the application of low dose constraints may be too restrictive.

3.3.2. General Implications for the Control of Radioactive Waste

The ICRP recommendations described above, seem to justify the following general radiological protection framework for the control of radioactive waste.

Assessed doses and risks to appropriate critical groups provide an adequate basis for evaluating the protection being afforded future generations. Collective doses play only a minor role even in the process of optimization of protection.

The assessed doses and risks can be regarded as measures of health detriment for time periods up to several hundreds of years into the future only. Beyond this time they should be regarded as indicators of safety.

Assessments being undertaken to evaluate the degree of protection being afforded future generations should consider two broad categories of exposure situations: natural processes and inadvertent human intrusion. The term 'natural processes' covers all processes leading to exposure of humans other than human intrusion, i.e. pathways where human action has degraded one or more of the containment barriers.

Doses and risks assessed for natural processes should be compared with the appropriate constraints for protection of the public from controlled practices. However, it is inappropriate to compare doses assessed for human intrusion with constraints developed for application in controlled practices because for inadvertent human intrusion to have occurred, control over the repository must have been lost, at least to some degree.

The significance of doses assessed for inadvertent human intrusion can be evaluated by comparison with the range of doses experienced in various situations today. This approach is suggested instead of ICRP's approach of linkage to intervention criteria, to avoid possible misunderstandings that could arise from the simultaneous use of practice criteria and intervention criteria (in fact, the approach adopted in this report and ICRP's approach result in the same numerical criteria). Annual doses from natural background provide a useful barometer against which to judge the significance of doses. Typically on a worldwide basis, annual doses from natural background range up to about 10 mSv with a maximum in a few small areas of about 100 mSv. If the assessed dose for a particular disposal system is lower than around 10 mSv/a, the radiological consequences of human intrusion can be considered to be broadly tolerable provided there are no straightforward means for their reduction. If the assessed doses are higher than this value, consideration should be given in the design of the repository system to reduce either the probability of intrusion (by, for example locating the repository at a greater depth) or its consequences (by, for example, dilution of the waste). The target is to reduce intrusion doses to below 10 mSv per year. If the assessed dose is greater than 100 mSv then it must be established that all reasonable steps have been taken to mitigate the implications of human intrusion.

The process for achieving an acceptable level of protection of human health in the future from the management of radioactive waste should be constrained optimization with emphasis on taking all reasonable steps to achieve protection instead of relying solely on compliance with numerical criteria. Furthermore, the fact that specified numerical criteria may be assessed as being exceeded at some time in the future may not in itself imply rejection of the disposal option: the decision making process is judgmental particularly when considering the implications of doses assessed for time periods greater than several hundreds of years into the future.

3.3.3. Implications for the Interpretation of the Principles of Radioactive Waste Management

The new recommendations from ICRP focus on protection of the public; other issues such as protection of the environment, transboundary implications, the implications for national legal frameworks and matters relating to waste generation are not specifically addressed. Therefore, these new recommendations are of most relevance to the interpretation of principles 1, 4 and possibly 5 of the safety fundamentals (see Section 3.1). The main difficulty concerns the interpretation of principle 4 which is addressed at the protection of future generations and so this will be discussed first.

As stated in section 3.1, principle 4 is commonly interpreted as meaning that the same numerical criteria (limits and constraints) as are applied today should be applied at all times in the future. The implications of the ICRP recommendations are that for natural processes, dose projections in the far future indicating doses in excess of these levels should not necessarily be seem as contra indicators to acceptability. This is 'justified' by ICRP on the basis of the conservatism inherent in the calculations and so does not conflict greatly with the conventional interpretation of this principle.

It is in the consideration of inadvertent human intrusion that more serious issues arise. The conventional approach to protection of future generations in this context has often been to assess annual risks from intrusion, taking account of the assessed probability of such intrusion and of the corresponding health effects, and to compare the result with a risk constraint equivalent to the corresponding dose constraint. However, there is no consensus in this regard. The ICRP approach is to look at the consequences of intrusion in terms of dose. In judging the acceptability of this assessed dose ICRP refers to criteria for intervention on the basis that control over the source has been lost. If uncontrolled exposures resulted from a source today, the situation would be addressed using the system of protection for intervention and so it can be argued that this approach provides an appropriate level of protection for future generations. However, in applying this recommendation to a near surface facility for long lived waste it could be argued that as control is almost certain to be lost at some time in the future and so intrusion is almost certain to occur, essentially intervention criteria are being applied to the control of a practice. There is no clear answer to this issue and differences in interpretation seem to be unavoidable as discussed further in section 5.

Turning to Principle 5, which concerns undue burdens on future generations: the implications of the ICRP recommendations are that institutional control should not necessarily be regarded as an undue burden on future generations. This possibly is in accord with current thinking in this area although as was pointed out in section 2, the main issue is for how long one can rely on controls and not whether they are an undue burden or not. Concerning Principle 1 — securing an acceptable level of protection for human health — there do not appear to be any implications from the ICRP recommendations other than those noted under Principle 4 insofar as the two principles overlap.

3.3.4. Application to Disposal of Longer Lived Waste in Near Surface Facilities

These principles could be applied to the specific case of disposal of longer lived waste in near surface facilities, particularly mining and minerals processing waste in the following way. Any chosen option should comply with the dose or risk constraints for natural processes. This may require containment barriers to be monitored and at some stage in the longer term repaired. Facility designs should be robust and simple to repair in order not to place undue burdens on future generations but nevertheless the implications are that to achieve the level of safety required in the longer term, institutional controls will be necessary. The time period over which institutional controls are likely to remain effective cannot be guaranteed although every effort should be directed at their maintenance on an ongoing basis.

Assessed doses from human intrusion are likely to be important determinants in deciding whether waste are suitable for surface or near surface disposal. Appropriate stylized intrusion scenarios should be selected. For surface or near surface disposal of mining and minerals processing waste these are likely to include consideration of exposure to radon in dwellings constructed on or near the repository. In circumstances where the assessed doses are greater than 100 mSv/a modifications should be made to the repository concept to reduce either the probability of intrusion or its consequences. If the high doses are from radon exposure in dwellings, this could be achieved by either diluting the waste within the repository or disposal at greater depth.

If the assessed doses from intrusion are less than 100 mSv per year, it should be demonstrated that mitigation cannot be achieved without disproportionate use of resources by applying optimization procedures. If the assessed doses are less than 10 mSv/a, this requirement is relaxed and a simple check that straightforward measures will not bring about improvement is all that is required. Thus, an annual dose of 10 mSv can be viewed as a target in the optimization process in respect of intrusion considerations.

The radiological protection framework outlined above in respect of near surface disposal of longer lived waste is consistent with new ICRP recommendations. However, it may have significant implications in respect of mining and minerals processing waste. It implies that ongoing institutional control will be necessary and it implies that mining and minerals processing waste above certain concentrations would be contra indicated for near surface disposal or would require dilution to levels that could not give rise to intrusion doses above 100mSv. In supporting such an approach comparison with situations giving rise to exposures from naturally occurring background radiation may be helpful.

4. ISSUES IN ESTABLISHING A COMMON FRAMEWORK

4.1. General issues

There are several issues of a general nature which may influence the choice of which option may be appropriate to dispose of a particular waste type. The following sections discuss these and provide some general understanding of how these issues may affect the decision-making process. Further details will be provided in the specific discussion of each waste type in relation to each disposal option.

4.2. Institutional control

The role of institutional control is to reduce the probability of intrusion into disposed waste, to reduce the magnitude of the consequences if intrusion does occur, to expedite intervention activities after intrusion has taken place and to help achieve societal confidence. Monitoring and inspection are particular forms of institutional control and are very important part of generating societal confidence. The term “institutional control” refers to that period after “normal” operation of a repository has ceased even though there is continuous involvement of institutions during all stages of any waste facility.

The half lives of radioisotopes can be clustered into groups of: short lived — less than 1 year, intermediate lived — up to about 30 years and long lived — typically a few thousands of years or more. Ten half lives decay reduces the source activity concentration by a factor of one thousand, for large sealed sources, twenty half lives gives a reduction of a million times. Thus an institutional control period of 300 years would allow most radioactive waste to decay to trivial levels. The exceptions being large sealed sources, high level waste/spent fuel and naturally occurring isotopes. For very long lived radioactive waste such as that containing the primordial naturally occurring radioactive species in the uranium and thorium decay chains no reasonable period of decay would allow activity levels to decay significantly.

The concept of institutional control changes over time. The “practice” period could lasts up to 50 years or more and includes the operating period and the closure phase. Post closure starts after this but there can be an intermediate period referred to as the Post Operational Pre-closure Phase (POPC) during which extended monitoring can be undertaken to ensure facility performance is satisfactory. At some point there is a shift in emphasis and this may also be reflected in a change in ownership and the facility (usually to a government body) as well as a change in the degree of regulatory control.

For near surface disposal facilities containing low and intermediate level waste of short or intermediate half life and limited amounts of longer lived waste, it would seem reasonable from a radiation protection perspective to rely on institutional controls to achieve safety.

Generally this has been considered to be reasonable for a limited period of time (up to several hundred years) and also for this sort of time period that institutional controls can also be thought of as additional or complementary safety barriers which work with the other natural and engineering barriers to ensure safety. Institutional controls do not constitute an "undue" burden from a radiation protection perspective. They are a burden, but are seen by society as acceptable in the context of managing other types of hazardous chemical waste. The important issue, and maybe burden, is that the current generation pass on to the next generation the knowledge, skills, records and societal judgements that led to the existing decisions as well as any financial resources needed to cover work that was intentionally deferred. This would allow the next generation to make the decisions it regards as being appropriate and acceptable to it. This might include stopping any further action, reversing past actions or continuing to pass information on to its immediate next generation. It is not possible, reasonable or practicable for this generation to impose its will on future generations. It is the responsibility of each generation to consider, to decide and to act. Thus there is perhaps limited value in a philosophical discussion on what might happen in the next few hundreds or thousands of years.

Regarding geological disposal facilities institutional controls are not regarded as being necessary to ensure safety. They are complementary to other barriers but could help to build societal confidence. Radiological monitoring is undertaken to facilitate societal confidence as there are no expected consequences that can be observed for very long times. Markers and passive land use controls may be considered to be appropriate and passing of records and other design and decision-making information should be carried out. Safeguards will need to be maintained for spent fuel or other fissile material as determined.

For long lived and naturally occurring materials, institutional control of near surface disposal facilities is necessary to prevent intrusion into the waste and to prevent diversion of the waste – an issue often of concern for mine tailings and waste rock dumps. There is no limit to the time period where this is needed, it is analogous to the way society deals with long lived chemically hazardous waste. Robust mechanisms are needed to ensure that transfer of information between generations is maintained. Where practicable, such radioactive waste should be disposed far enough bellow the surface to avoid the need for open ended institutional controls. Monitoring/ inspection are important to identify barrier degradation and to undertake repairs as needed and to determine if there are direct effects on the immediate environment and to allow prompt intervention if intrusion occurs.

Safeguards are a specific form of institutional control. They apply only to spent fuel where the amount of fissile material is above the level considered to be practically irrecoverable under the Non Proliferation Treaty. It would also apply to weapons grade plutonium if it were to be considered as waste and placed into a repository. The key issue for safeguarding waste is to ensure that no measures that are taken to verify the materials significantly compromises the overall safety of the repository. Thus safety has priority over safeguards. This usually results in maintaining the integrity of the barrier systems and thus safeguards activities are proposed which rely on remote sensing and site monitoring. As there is currently technology available to do this, having a safeguards regime in place does not impose any special constraint of the choice of how such waste is disposed.

4.3. Intrusion

Intrusion can be defined as human actions that are not intentional which give rise to consequences from a waste repository or other form of waste management facility. Intentional intrusion is specifically excluded. In order to determine what effect intrusion may have on the choice of disposal option for any given waste, it is necessary to first consider some related aspects such as: how to evaluate the consequences which may arise in the future from human intrusion, how to judge the acceptability of the consequences which might arise given that they will have a probabilistic nature (they are not certain to arise) and that they may have a magnitude that would not be acceptable if they where to occur in a normal practice situation today, how to reduce both the likelihood of the intrusion occurring and the magnitude of the consequences if it does?

In terms of evaluating the consequences, given the long time frames and the associated uncertainties, only an indication of consequence is reasonable and appropriate. It would appear to be reasonable to use a few generic stylized scenarios as a basis for indicating what the consequence might be. Scenarios should be somewhat conservative , but not unreasonably so. This approach is somewhat analogous to the use of "reference man" in the radiation protection context. It is generally considered that analysis should not be overly complex since that would imply an unreasonable level of precision and understanding in the light of future changes in social habits and biosphere evolution. Three basic intrusion situations have been suggested: direct access and immediate exposure to waste giving rise to high doses (driller being exposed to spent fuel from a core sample), direct access to waste and prolonged exposure to low level doses (waste is removed from the repository and people live on or near the waste.) and indirect exposure to low doses from a leaking repository whose barriers have been degraded by the intrusion. Exposures are received sooner than expected from natural evolution processes. (a well penetrates a contaminant plume from a degraded barrier system due to the intrusion).

In terms of judging acceptability of the consequences the approach recommended by ICRP can be adopted. The thinking infers doses are potential in nature and would occur at a time when the waste repository is no longer a "practice", but rather would be considered an "intervention" if it where to occur today. Doses would be estimated from the three generic stylized scenarios as well as their probability (the dis-aggregated approach). Doses above 100 mSv should not be allowed and every effort be made to reduce the consequences or the likelihood of occurrence. Doses of less than 10 mSv would likely be acceptable if all practicable and reasonable measures have been taken to reduce/optimize exposures or their likelihood. Doses less than 1 mSv would be acceptable and require no further action.

There are a number of possibilities to reduce both the likelihood of intrusion and the possible consequences. The likelihood can be reduced by increasing the depth of disposal, adding markers to ensure that any intrusion sees "made ground" and by maintaining records and other passive institutional controls. The consequences could be limited by elimination of hot spots in the repository, by dispersing the waste in the repository to reduce the specific activity or by selecting a site with inherently good containment capacity to reduce dependency on engineered barriers which could be penetrated.

4.4. Reversibility and retrieveability

The benefit of having a waste repository which can have one or all of its various operations reversed is that less emphasis has to be placed on having to demonstrate that all aspects of the design are adequate from the outset. The knowledge that the disposal process can be reversed could help generate confidence on the part of the various interested parties (stakeholders) within a society, although it should not relieve the developer or operator of the repository of the obligation to demonstrate with a high degree of likelihood that the repository will be safe.

It is not surprising that society is not generally willing to accept a waste facility which cannot be repaired and which it is claimed does not need to be repaired. This runs counter to human experience where failures occur and mistakes are made on a regular, if not frequent basis. Claims that the understanding of repository systems and their behaviour over long periods at the early stages of development have little credibility and can antagonise public sentiment.

For any short lived waste, it is unlikely that reversibility is an issue as they will decay in time periods that would allow the waste to be left in place even if it were not found to have been placed in an optimum situation. Thus reversibility will only be an issue for waste that are long lived such as spent fuel, high level waste and uranium mine tailings. Of these, waste placed on or near the surface are inherently very accessible and are thus inherently retrieveable without special measures being needed to enhance or facilitate access. Thus in general the only waste that need to be considered specifically from a reversibility perspective is long lived waste placed into a deep geological repository.

The key question is whether measures which are taken to facilitate reversibility are likely to degrade barrier systems used to isolate and contain the waste and thus to reduce the safety of the facility. It is generally considered that safety should be the primary consideration. Studies have shown that the safety can be maintained even when provisions are made to enhance access to the waste to facilitate reversibility. In addition the cost of such provisions is not usually very significant when compared to the overall cost of a repository. The main constraint which these studies reveal is that good institutional controls must be maintained to ensure that the waste is not allowed to flood and that the repository is ultimately closed in accordance with its initial design. If either of these conditions are not met, then there is a likelihood that a significant reduction in safety can occur. This is seen to be unacceptable and thus any delay in closing a repository should be reduced to the minimum and be accompanied by a strong institutional control framework. There is however no fundamental reason from a safety perspective not to allow for reversibility and ultimate recovery of waste placed in a disposal repository.

4.5. Making the safety case

Whilst it is generally recognised that quantitative performance and impact assessment is important in the design of disposal facilities, in establishing operational controls and limitations and in associated regulatory processes, it is not enough to provide the levels of assurance necessary in the overall safety of facilities and the associated decision making processes both regulatory and at a broader political level.

A safety case should be developed for a repository which addresses the logic and soundness of the overall system design and a range of aspects which provide confidence in the design and the more quantitative aspects of safety impact assessment. It should address the robustness of the design explaining the passive safety features adopted, the mechanisms that will provide for containment of the waste and for isolation form the accessible environment. It should address the reliability of the various natural and engineered design features and their assessed performance. The safety case should explain how the concept of defence in depth has been employed and what features provide multiple barriers or levels of safety. It should also address the safety margins designed into the facility. The integrated performance assessment of the repository should feature in the safety case which due consideration of uncertainties. The adequacy, appropriateness and validation of the modelling process and models used in the assessment should be addressed. The level of confidence associated with the safety of the repository should be addressed as relevant to a particular stage of the repository development.

Aspects such as natural analogues, alternative indicators of safety to dose and risk and monitoring should be included.

5. APPLICATION OF A COMMON FRAMEWORK

The basic reason for developing a common framework for disposal of radioactive waste is in recognition of the fact that a range of radioactive waste types exist and are continuing to be produced, the majority of which need to be disposed of in order to ensure compliance with the internationally agreed waste safety fundamentals. A number of disposal options are available and have been utilised in the past. In reviewing past waste disposals practices and when looking at proposed disposal practices for the future, there are clearly some areas where anomalies exist. There is a clear need to find safe and cost effective disposal solutions for spent radiation sources in countries which are not engaged in nuclear programmes to any significant degree or with limited infrastructure and resources. Borehole disposal could be such an option. There is also a need to rationalise the near surface disposal of mine tailings with the accepted waste disposal principles and to identify acceptable options for disposal of NORM waste.

The coming into force of the Joint Convention on the safety of Spent Fuel and the Safety of Radioactive Waste management will no doubt raise the focus on these and related issues. The development of an internally endorsed common framework for disposal of all radioactive waste types will assist in decision making in this area particularly is such a framework can enable a clear and justifiable safety argument to be developed for the disposal of all types of radioactive waste in cost effective, safe and appropriate disposal facilities.

Such a common framework must for each waste type, processing and disposal option systematically test all the waste and radiation safety principles and determine congruence and compatibility.

6. CONCLUSION

The principles of radioactive waste safety are now sufficiently developed to structure a framework against which waste types and disposal options can be tested. A systematic mechanism needs to be developed for structuring and applying such a framework.

SAFETY OF RADIOACTIVE WASTE MANAGEMENT IN FRANCE

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Abstract. Radioactive waste produced in France vary considerably by their activity level, their half lives, their volume or even their nature. In order to manage them safely, the treatment and final disposal solution must be adapted to the type of waste considered by setting up specific waste management channels. A strong principle in France is that it is the responsibility of the nuclear operators as waste producers to dispose of their waste or have them disposed of in a suitable manner. The competent authorities regulate and control the radioactive waste management activities. At present, only short-lived low and intermediate level waste have a definitive solution, the surface repository, where adequate waste packages are disposed of in concrete structures. Other types of radioactive waste are in interim storage facilities at the production sites. For very low level waste coming mainly from dismantling of nuclear facilities a dedicated repository is planned to be built in the coming years. Dedicated repositories are also planned for radiferous, tritiated and graphite waste. As for high level waste and long-lived waste coming mainly from reprocessing of spent nuclear fuel the disposal options are being sought along the lines specified by law 91-1381 concerning research on radioactive waste management, passed on December 30, 1991:

— research of solutions to partition and transmute long-lived radionuclides in the waste;
— studies of retrievable and non retrievable disposal in deep geological layers with the help of underground laboratories;
— studies of processes for conditioning and long term surface storage of these waste.

In 2006, the French Parliament will assess the results of the research conducted by ANDRA relative to deep geological disposal as well as the work conducted by CEA in the two other areas of research and, if this research is conclusive, pass a law defining the final disposal option.

1. NATIONAL FRAMEWORK

1.1. National policy

The French nuclear activities produce solid, liquid or gaseous waste, some of which is radioactive. The national policy on radioactive waste is that reliable, transparent and stringent management of this waste must ensure the protection of individuals, preservation of the environment and limitation of undue burdens imposed on future generations.

Concerning the control of the safety of nuclear activities, including waste management, an extensive regulatory system has been set up consisting of laws, decrees and guidance rules. Its objective is to ensure the safety of nuclear facilities and the protection of man and environment. The regulation concerning radioactive waste management deals with, waste management strategies, design basis related to safety, quality, release of effluents, impact on the environment, radiation protection, performance assessment, licensing procedures, incidents, the organisation of the control of safety, involvement of the public.

A strong principle in France is that it is the responsibility of the nuclear operators as waste producers to dispose of their waste or have them disposed of in a suitable manner. The competent authorities regulate and control the radioactive waste management activities.

A specific public agency, ANDRA, has the responsibility for the long term management of radioactive waste. This agency operates waste repositories, defines the acceptance criteria for waste packages in these repositories and controls the quality of their production. It also keeps a national inventory of radioactive waste in France.

In order to share the experience of other countries, France signed the "Joint convention on the safety of spent fuel management and on the safety of radioactive waste management" on September 5, 1997. The convention was ratified on April 27, 2000.

1.2. Institutional framework

The French Nuclear Safety Authority is entrusted with the definition and application of the regulations of the main nuclear facilities, known as “basic nuclear installations” (BNI’s) such as nuclear reactors, fuel cycle plants, shut-down nuclear facilities, waste treatment plants, radioactive waste interim storage facilities and final repositories. It is under the supervision of the Ministry of economy, finance and industry and the Ministry of territorial management and environment.

The French Nuclear Safety Authority combines the resources of the Nuclear Installations Safety Directorate (DSIN), the NSSS control office (BCCN) and the Nuclear Installation Departments (DIN) set up within the Regional Directorates for Industry, Research and the Environment (DRIRE).

The main assignments of the French Nuclear Safety Authority are:

— drafting and monitoring the application of the general safety regulation;
— implementing licensing procedures for BNI’s;
— organising and implementing surveillance of BNI’s.

Nuclear facilities which are not considered as Basic Nuclear Installations because they deal with a quantity of radioactive material at an activity level below the threshold of BNI’s are required to comply with the environmental protection provisions specified in the law 76-663 of July 19, 1976, insofar as they belong to the category of facilities classified on environmental protection grounds (ICPE). They are controlled at the local level by the DRIRE under the supervision of the Ministry of Environment.

Uranium mines and mills are respectively under the supervision of the Ministry of Industry and the Ministry of Environment. Nuclear installations connected to military activity are under the control of the “Haut Commissaire à l’Energie Atomique”.

As for the research relative to the management of high level and long-lived waste, the law of 30 December 1991 prescribes that a National Review Board be created, audit the different actors of this research and publish a report to the government each year.

1.3. Regulatory framework

In the framework of its regulatory functions, DSIN issues basic safety rules (RFS) which constitute guidelines defining the safety objectives to be achieved and describing accepted practices deemed compatible with these objectives.

The licensing of BNI’s is performed within the framework of the decree of December 11, 1963 which provides for an authorisation decree procedure followed by a series of licences issued at key points in plant lifetime: provisional license for start-up of normal operation, definitive license after several years of operation, decommissioning licenses.

Before the authorisation decree is signed, the facility has to provide a preliminary safety analysis report and an environmental impact study. The reports are subjected to public debate in the framework of a public inquiry (law 83-630 of 12 July, 1983 and decree 96-388 of May 10, 1996). A technical instruction procedure is followed implying a peer review by an advisory committee (the standing group of experts). Consultations of the different ministries concerned are set up.

2. CURRENT STATUS OF WASTE MANAGEMENT

2.1. Waste classification

Radioactive waste produced in France vary considerably by their activity level, their half-lives, their volume or even their nature (scrap metal, rubble, oils, etc.). The treatment and final disposal solution must be adapted to the type of waste considered in order to manage them safely.

The radiological risk can be assessed on the basis of two main parameters: the activity level, indicating the toxicity of the waste, and the half-life, which depends on the radioactive decay periods of the radioelements it contains.

The classification makes the distinction between short-lived waste and long-lived waste, and on the other hand on the distinction between very low, low, medium or high level waste. It is based on the existing or expected management channels.

Table I. Existing or future disposal channels for the main solid waste and residues resulting from radioactive waste treatment

<table>
<tr><th>Activity/period</th><th>Short-lived</th><th>Long-lived</th></tr>
<tr><td>Very low level</td><td>Dedicated disposal facilities (under investigation)
Recycling channels (under investigation)</td><td>Conversion of current storage areas into disposal facilities (under investigation)</td></tr>
<tr><td>Low level</td><td rowspan="2">Surface disposal at the Aube repository
Recycling of certain metals (under investigation)
Dedicated disposal facilities for tritiated waste (under investigation)</td><td>Dedicated disposal facilities planned for waste containing radium and graphite (under investigation)</td></tr>
<tr><td>Medium level</td><td>Waste management channels being devised in the framework of law 91-1381 of December 30, 1991</td></tr>
<tr><td>High level</td><td colspan="2">Waste management channels being devised in the framework of law 91-1381 of December 30, 1991</td></tr>
</table>

2.2. General waste management strategy

For the French Nuclear Safety Authority, the management strategy adopted must cover all categories of waste. This involves setting up specific waste management channels, taking into account not only radiological risks, but also chemical and sometimes biological hazards incurred by these waste.

The waste management begins with the nuclear plant design, proceeds during the operating life of the installation through concern for limitation of the volume of waste produced, of its noxiousness and of the quantity of residual radioactive materials contained. It ends with waste elimination (recycling or final disposal) via the intervening stages of identification, sorting, treatment, packaging, transport and interim storage. All operations associated with management of a category of waste, from production to disposal, constitute a waste management channel, each of which must be adapted to the type of waste concerned.

The operations within each channel are interlinked and all the channels are interdependent. These operations and channels form a system which has to be optimized in the context of an overall approach to radioactive waste management encompassing safety, traceability and volume reduction issues.

The objective of the French Nuclear Safety Authority is to ensure that all categories of radioactive waste are managed safely and find an outlet.

Table I shows the stage reached in implementation of the different waste management channels, notably the final disposal channel adopted. The absence to date of definitive disposal solutions for certain categories of waste will be noted.

2.3. Waste management adapted to the different categories of waste

2.3.1. Very low level waste

Very large quantities of very low level waste were produced in the past during operation of the French uranium mines. These waste contain a very small quantity of long-lived radionuclides, notably radium. Since moving the millions of tons concerned is obviously out of the question, it is planned to restructure the mining sites where they are currently stored, taking maximum advantage of the characteristics of these residues (low solubility and permeability), aiming at a long term stable solution, requiring neither frequent maintenance nor constant surveillance.

Today's very low level waste come mainly from the dismantling of nuclear facilities or conventional industrial sites using slightly radioactive substances. The quantity involved will increase considerably when the time comes for the large scale complete dismantling of power reactors currently in operation. Radioactivity in these cases amounts to a few bequerels per gram.

The very low level waste are currently temporarily stored on production sites. Efforts are being made to rationalize their management. The approach developed in this context falls within the scope of law 75-633 of July 15, 1975 concerning waste disposal and recovery of materials. The basic principle of this law are the responsibility of the waste producers, the traceability of this waste and the necessity to inform the general public

On December 31, 1999 a ministerial order dedicated to waste management in nuclear facilities was signed. Operators of nuclear facilities should produce "waste studies" which describe the management of the different categories of waste in their facilities and the corresponding disposal channels.

ANDRA and France-Déchets announced in 1999 the setting up of a partnership to create a dedicated VLLW repository next to the Aube Disposal Centre. Three potential zones have been selected. Following this survey phase, the characterization phase has begun. The facility may be operational in 2004.

2.3.2. Long-lived low level waste

The long-lived low level waste include the particular category of waste containing a significant quantity of radium and producing radon. These waste were notably produced in the past by the rare earth industry. A special method of disposal is currently being studied by ANDRA for this type of waste.

2.3.3. Short-lived medium and low level waste

The activity of short-lived medium and low level waste, designated "A waste" by the nuclear operators, is mainly due to beta or gamma radiation emitting radionuclides, with a half-life of less than 30 years. Alpha particle emitters are strictly limited.

Short-lived low-level or intermediate-level waste (LILW) are generated by nuclear reactors, fuel cycle facilities, research centres, as well as university and hospital laboratories (15,000 m^3/year). They consist mainly of manufacturing waste, worn equipment, materials cleaning rags and protective clothing. This category also includes products from gaseous and liquid waste treatment at nuclear installations.

The reference technical solution adopted for the long-term management of this type of waste in France is disposal in a surface repository where adequate waste packages are disposed of in concrete structures.

This provides for containment of the radionuclides during a sufficient length of time for their activity level to decay. In the past, this type of waste was disposed of at the Manche Disposal Centre. It is disposed of, since 1992, at the Aube Disposal Centre. These two Centres are operated by ANDRA.

For surface disposal, an important issue is the definition of the conditions for entering the surveillance phase. With regard to the Manche Disposal Centre (surface repository), which operation ended in 1994 and which is now set to enter the surveillance phase, the French Nuclear Safety Authority makes sure that the recommendations laid down by the committee on the subject set up in 1996 by the Ministers for Industry and for Environment are applied. The French Nuclear Safety Authority has formally approved the safety report produced by ANDRA in January 1999. A public inquiry was held from February 2nd to April 3rd 2000 as part of the licensing procedure for entering the surveillance phase. It concerns as well the revision of the ministerial order for authorization of effluent release relative to this facility. The decree creating the new facility in the surveillance phase should be signed in 2001.

The Aube Disposal Centre was authorized by a decree of September 4, 1989. Its lifetime is planned for 30 years and could be extended to 60 years as the quantity of waste yearly received has drastically decreased. Following the provisions of the creation license, issued in 1989, ANDRA sent the final safety report for the repository to the French Safety Authority, in December 1996, integrating operating feedback from the first years of operation. On this basis the definitive operating license was granted on September 2nd 1999. In parallel, the authorization for effluent releases is in the process of being revised. In 2000, 13,240 m^3 of LILW have been delivered to the Aube Centre and three structures have been closed. Since the beginning of its operation the Aube Centre had received, at the end of 2000, 111,6551 m^3 of LILW.

In May 1995, the French Nuclear Safety Authority defined, in the basic safety rule RFS III.2.e, revised requirements for radioactive waste package acceptance for disposal in a surface repository. The respective responsibilities of ANDRA and the waste producers are precised in this rule. The Safety Authority carries out inspections to check that the acceptance procedures comply with the requirements of RFS III.2.e and are correctly implemented. Taking into account the experience feedback and the final safety report of the Aube Disposal Centre, the RFS III.2.e is currently in the process of being revised.

A new facility, CENTRACO, has been licensed to operate at the beginning of 1999. It receives short-lived low level or intermediate level waste either for incineration or, in case of metal scrap, for melting. It contributes to minimization of the volume of the waste before its disposal in a surface repository.

2.3.4. High level waste and long-lived medium level waste

Long-lived radioactive waste containing alpha emitters are divided into medium level and high level waste. The former category, known as “B waste” by the nuclear operators, derives mainly from the process, the operation and maintenance of reprocessing facilities (1600 m^3/year). The latter category known as “C waste” stems from fission and activation products arising from the reprocessing of spent fuel (240 m^3/year). These high activity waste also include CEA fuel, irradiated in research reactors and currently unused, together with EDF spent fuel which is not intended for reprocessing.

Disposal options for this type of waste, currently stored on the production sites, are being sought along the lines specified by law 91-1381 concerning radioactive waste management, passed on December 30, 1991. This law requires the implementation of a fifteen years research programme along three different areas of research:

— research of solutions to separate and transmute long life radionuclides in the waste;
— studies of retrievable and non retrievable disposal in deep geological layers with the help of underground laboratories;
— studies of processes for conditioning and long term surface storage of these waste.

The discussion and information process conducted by the mediator Christian Bataille, member of the French parliament, and the favorable geological characteristics, led to the choice in January 1994, by the government, of four geological areas in the departments of GARD (clay), VIENNE (granite), HAUTE-MARNE (clay) and MEUSE (clay).

Preliminary surface investigations carried out by ANDRA allowed this agency to select three potential sites for the location of a deep geological laboratory. One is located at the border between the two departments of Meuse and Haute-Marne and is now called East site. The two others are located in Gard and in Vienne.

In June 1996, the government allowed ANDRA to apply for the creation of laboratories in these three sites. ANDRA applied for the East, Vienne and Gard sites mid 1996. DSIN sent the application to the Prefets of the Departments who organized local public inquiries and asked for advice of the local administrations, as well as elected representatives. At the same time, the applications provided by ANDRA were submitted to a review by IPSN and the standing group of experts on waste management in March and April 1997, on the basis of the basic safety rule, RFS III.2.f, issued in 1991, which defines the general safety objectives for geological disposal of high level waste and long-lived waste. On the basis of the reports from the Prefets and the conclusions of the review, DSIN considered in its report of December 1st, 1997 that two sites were suitable, the East site and the Gard site.

The French government decided, on December 9th, 1998, that the research on geological disposal should be performed at two sites: a clay site at Bure (East France) and a granitic site to be selected.

After the research program has been performed, one of the selected site for a deep geological laboratory could be proposed to the parliament as the location for a waste repository.

Following this decision, the government signed, on August 3rd 1999, three decrees:

— the decree authorizing ANDRA to implement and operate at the Bure site, in Eastern France, an underground laboratory in order to study deep geological formations where radioactive waste could be disposed of;
— the decree giving general guidelines to set up local committees in charge of following the activities implemented in underground laboratories;
— the decree deciding the creation of a commission composed of three members in charge of the dialogue prior to selecting one or several granitic sites for a second underground laboratory.

Concerning the first decree, the French Government granted, on August 7th 2000, the authorization to ANDRA to sink the shafts of the underground laboratory. The sinking of the shafts is underway since September 2000 and will last for two years. Further ministerial authorizations will be required for drilling the drifts of the underground laboratory.

Investigations and experiments in the underground laboratory should allow to gather the necessary data for the conception, the optimization (with respect to retrieavability) and the demonstration of the safety of a potential repository on the site.

Prior to the August 7th, 2000 authorization, ANDRA sent to the French Nuclear Safety Authority a series of documents concerning the geology of the Bure site, the initial design option for a repository, the safety approach and the experimental program during shaft sinking. ANDRA sent as well a global development plan concerning the research relative to the deep disposal project.

On the request of the French Nuclear Safety Authority, these documents were analyzed, during the year 2000, by the Nuclear Protection and Safety Institute (IPSN) and the Standing Group of Experts on waste management.

This resulted in a series of recommendations that the French Nuclear Safety Authority addressed to ANDRA for the following steps it conducts in deep geological disposal, i.e.:

— a first performance assessment
— a preliminary design concept
— an updated experimental program

Concerning the second decree, the local information follow-up committee was set up on November 15th, 1999.

Concerning the research of a granitic site, ANDRA set up an expert group composed of French and foreign experts and proposed, in 1999, a first selection of 15 sites on the basis of a bibliographical study. This first selection was approved by the National Evaluation Commission. The Government nominated on November 19th, 1999 the three members of the commission in charge of the dialogue on the selected sites. The commission began its work at the beginning of the year 2000 but had to stop because of a strong local opposition. A report to the Government was published on July 2000.

As set out by the law of December 1991, a National Review Board is in charge of evaluating the progress made in the three ways of research. Six reports were provided to the government and the parliament in July 1995, June 1996, September 1997, October 1998, July 1999 and June 2000.

In 2006, the French Parliament will assess the results of the research conducted by ANDRA in deep geological disposal as well as the work conducted by CEA in the 2 other areas of research.

3. CURRENT ISSUES/PROBLEMS

3.1. Waste inventory

In 1999, in order to improve the coherence of waste inventories, Mr. Y. Le Bars, President of ANDRA, was entrusted by the government with a mission of defining a methodology for the waste inventory from nuclear facilities. The report on this methodology was published on June 14, 2000. ANDRA thinks of being ready to publish a first detailed inventory in the course of the year 2003. Uncertainties will be clearly identified.

3.2. Tritiated, graphite waste

The short-lived medium and low level waste include certain categories which have characteristics making them unsuitable for acceptance at the Aube repository. These are waste containing tritium, which is difficult to confine, and also graphite waste, which contain a non-negligible proportion of long-lived radionuclides. A working party, led jointly by the High Commissioner for Atomic Energy and the Director of the DSIN, in which took part ANDRA and the various waste producers (CEA, COGEMA and EDF), is entrusted with devising the most suitable management channels for these types of waste. ANDRA is studying dedicated repository designs for these particular types of waste.

An anticipated regulatory issue will be the licensing procedures concerning the creation of disposals for specific types of waste. Guidance rules should be issued on that subject.

3.3. Surface disposal

For short-lived low and intermediate level waste the main concern is the regulation concerning the different phases of the institutional control period.

3.4. Interim storage of radioactive waste

For all waste for which a final solution has not been found, it is essential that satisfactory temporary solutions are implemented. The French Nuclear Safety Authority makes sure that these temporary solutions are not only safe, but also that they do not become definitive as a result of lack of action. In this respect, the CEA and COGEMA have started a clean-up of their installations where "old" waste are temporarily stored.

DSIN has initiated the elaboration of a new Basic Safety Rule concerning interim storage of radioactive waste. It is based on the experience feedback from existing interim storages.

3.5. High level waste and long-lived medium level waste

DSIN follows closely the research programs set up by ANDRA for the underground laboratories which should end up by the elaboration of a global assessment report in 2006.

In this process, DSIN is particularly concerned with:

- the priority that must be given to safety;
- the necessity to avoid delays and to respect the schedule of the law of December 1991;
- the need that the research developed in the laboratories be operational and not academic.

Following the recommendations addressed to ANDRA in June 2000, DSIN makes sure that a first iteration of an integrated performance assessment will be available early in order to check the adequation of the experimental program and the conceptual design options.

At the same time, DSIN ensures that the elaboration by ANDRA of waste package acceptance criteria for long-lived waste is progressing rapidly.

Two reflection groups have been set up in the framework of the French–German and French–Belgian collaboration between nuclear safety authorities in order to precise the methodology of performance assessment of deep geological repositories.

The topics cover:

- definition of radio-protection objectives in the framework of the revision of the ICRP 46;
- principles for repository design;
- elaboration and treatment of scenarios and in particular human intrusions;
- use of safety indicators;
- treatment of uncertainties;
- confidence building with multiple lines of reasoning.

At the same time the French Nuclear Safety Authority is working on new safety guidelines in order to provide evolution to the Basic Safety Rule RFS.III.2.f as well as on a new Basic Safety Rule on waste package design for long-lived waste. Regulatory guidance should be given as well on the research concerning long term interim storage.

With respect to the second area of research one difficulty for the regulator is the short period of time between the acquisition of the first results from the underground laboratory at Bure and the conclusions to be drawn in 2006, and presented to the government subsequently.

The compatibility of retrievability requirements and long term safety requirements is also an regulatory issue of concern.

BASIC STRATEGIES FOR RADIOACTIVE WASTE DISPOSAL IN JAPAN

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Abstract. In Japan, the basic strategies for safe disposal of radioactive waste generated from nuclear facilities have been examined separately. Examination of the common framework has been just started in this year by the sub-committee of the Nuclear Safety Commission (NSC) and will be finished within two years. This report describes the status of the basic strategies for each radioactive waste disposal and refers some issues concerned when making the common framework.

1. CURRENT STATUS OF BASIC STRATEGIES FOR RADIOACTIVE WASTE DISPOSAL

Current status of the basic strategy of disposal for each radioactive waste is shown in Table I.

Table I. Current status of basic strategies for radioactive waste disposal

Waste Categories		Atomic Energy Commission	Nuclear Safety Commission			Situation of establishment of regulatory law and provision
		Disposal scheme	Basic concept of safety regulation	Concentration limit	Examination guideline	
High-level radioactive waste		Examined (May '98)	1st report was examined (Sep '00)	–	To be examined	To be established
Waste containing TRU nuclides		Examined (Apr '00)	Under examination (June '00~)	To be examined	To be examined	To be established
Uranium waste		Examined (Dec '00)	To be examined	To be examined	To be examined	To be established
Waste generated from nuclear power plant	LLW containing comparatively high radioactivity	Examined (Oct '98)	Examined (Sep '00)	Examined (Sep '00)	Under examination (Nov '98~)	Almost established (Dec '00)
	Low-Level Waste	Examined (Aug '84)	Examined (Oct '85)	Examined (Feb '87~ Jul '92)	Examined (Jan '93)	Established (Mar '87~ Feb
	Very Low-Level Waste			Examined (Sep '00)	Under examination (Nov '98~)	Almost established (Dec '00)
Waste generated from radioisotope use, nuclear research and other related facilities		Examined (Jun '98)	Under examination (Jun '98~)	To be examined	To be examined	To be established
Clearance Level		Examined (Aug '84)	Under examination (May '97~)	Under examination (May '97~)	–	To be established

1.1. Waste category

Radioactive waste are separated into high-level radioactive waste (HLW) generated from a reprocessing plant (RP), waste generated from a nuclear power plant (NPP), waste containing TRU nuclides generated from RP and a mixed oxide (MOX) fuel fabrication plant, uranium waste generated from a uranium conversion and fuel fabrication plant and a uranium enrichment plant, and waste generated from radioisotope use, nuclear research and other related facilities. RPs include domestic plants and foreign RPs in France and UK where

Japanese electric power companies asked to reprocess a part of spent fuels. These waste are now stored at each facility and a part of them has been already disposed of at the disposal facilities as shown in Table II.

Table II. Amounts of radioactive waste being stored

Category of Waste		Cumulative Amount of Waste
High-level radioactive waste		464 canisters (vitrified waste) 850 ton of liquid waste [*1]
Waste containing TRU nuclides		87,000 drums (200L) at JNC[*2]
Uranium waste		112,000drums(200L) [*3]
Waste generated from nuclear power plant	LLW containing comparatively high radioactivity	8000 ton[*2]
	Low-Level Waste	500,000drums(200L) at Nuclear Power Plants[*4] 130,000drums(200L) were disposed at Rokkasho Disposal Facility
	Very Low-Level Waste	2000 t was disposed at JPDR site[*4]
Waste generated from radioisotope use, nuclear research and other related facilities		392,000drums(200L) [*2]

[*1] as of Feb '01.
[*2] as of Mar '98.
[*3] as of Mar '99.
[*4] as of Mar '00.

1.2. Basic strategies for radioactive waste disposal

(a) Waste generated from NPP

The disposal measures for NPP waste have been examined first. After the examination, these waste were separated into four categories of waste based on the radioactivity contained in them and their disposal concepts. They are clearance level waste (CLW), very low-level radioactive waste (VLLW), low-level radioactive waste (LLW), and LLW containing comparatively high radioactivity than LLW. VLLWs are concrete waste and metal waste, not solidified in a container and shall be disposed of in a shallow ground burial facility where no artificial structures are constructed. LLW shall be solidified in a container and shall be disposed of in a shallow ground burial facility where artificial structures are constructed. There are upper bounds of radioactivity concentration of LLW and VLLW, which can be disposed of in these facilities. Fundamental safety guidelines of land disposal of LLW and VLLW are shown in Table III.

Under regulation of "The Regulative Act for Nuclear Source Materials and Nuclear Reactor" (The Reactor Regulation Law), the disposal facility of LLW is now commercially operated by Japan Nuclear Fuel Limited (JNFL) at Rokkasho-mura in Aomori prefecture from 1992. The facility currently has a permission to allow for the disposal of 400,000 drums as a 200L drum of LLW from NPPs. Cumulative amounts of radioactive waste from NPPs are about 500,000 drums (200L) and about 130,000 drums (200L) have been already disposed of as of March in 2000. Japan Atomic Energy Research Institute (JAERI) operates the disposal facility of VLLW. To demonstrate the safety of VLLW disposal, about 2,000tons of very low-level concrete waste generated from Japan Power Demonstration Reactor (JPDR) are disposed of in the facility.

Table III. Fundamental safety guidelines of land disposal facility of LLW[*1]

(1) Disposal system by solidifying the waste in a container and disposing of the containers in shallow ground of waste burial facilities where artificial structures are constructed

Institutional control period				Post closure
Phase	First phase: Maintaining the integrity of the artificial barrier	Second phase: Securing the barriers performance	Third phase: Securing the barriers performance	
Concept of securing safety	Prevention of the leakage of radionuclides outside the artificial barriers and confirmation that there is no leakage.	Prevention of the effects of radionuclides reaching living environment by artificial barriers. Confirmation of safety by environmental radioactivity monitoring.	Prevention of the effects of radionuclides reaching living environment mainly by natural barriers. Prohibiting or restricting specific acts, i.e. digging out waste.	There is no necessity to take measure to prohibit specific acts.
Period	——	300y – 400y		——
Maximum dose standards	1mSv/y + ALARA			1010Sv/y for normal events Not exceed 10NoSv/y significantly for low probabilistic events

(2) Disposal system by disposing of nuclear concrete waste not solidified in a container in shallow ground of waste burial facilities where no artificial structures are constructed..

Institutional control period			Post closure
Phase	Burying phase: Securing the barriers performance	Conservation phase: Securing the barriers performance	
Concept of securing safety	Prevention of the effects of radionuclides reaching living environment. Confirmation of safety by environmental radioactivity monitoring.	Prevention of the effects of radionuclides reaching living environment by natural barriers. Prohibiting or restricting specific acts, i.e. digging out waste.	There is no necessity to take measure to prohibit specific acts.
Period	——	– 50y	——
Maximum dose standards	1mSv/y + ALARA		1010Sv/y for normal events Not exceed 10NoSv/y significantly for low probabilistic events

*1 "Fundamental Guidelines of Licensing Review of Land Disposal Facility of LLW", 1985.

According to "The Fundamental Guidelines of Licensing Review of Land Disposal Facility of Low-Level Radioactive Waste" (March 1988), the radiation dose to the public due to the buried waste after the completion of the institutional control period should be low enough to be able to diminish the management from the point of view of radiation. This dose refers to the level of radiation dose of 10 µSv/y issued as the exemption dose by the Radiation Council in its report "Exemption Dose for Shallow Land Disposal of Radioactive Waste" (December 1987). For the events with lower probability of occurrence, the value of assessment of the radiation dose should not too much exceed 10 µ Sv/y.

Regarding LLW containing comparatively high radioactivity than LLW, such as core internal metals, ion exchange resins, etc., the Advisory Committee on Nuclear Fuel Cycle Backend Policy (the Advisory Committee) of the Atomic Energy Commission (AEC) has examined the disposal strategy and published the report in October 1998. The Advisory Committee concluded that such kind of waste shall be solidified in a container and shall be disposed of in a rock cavern at the depth of around 50 m to 100 m where artificial structures are constructed. The depth means sufficient distance not disturbing usual usage of under-ground.

(b) HLW

In November 1999, the Japan Nuclear Cycle Development Institute (JNC) submitted a series of technical reports regarding the geological disposal of HLW to AEC. The overall intention of these reports, which together make up “The Second Progress Report on Research and Development for the Geological Disposal of HLW” (H12 report), was to demonstrate the safety of the geological disposal of HLW and to establish the scientific and technical basis for the geological disposal of HLW. The concept of the geological disposal of HLW in Japan is to ensure the long-term safety with a multiple barrier system consisting of robust engineered barriers in a stable environment. The engineered barriers are composed of HLW glass itself, overpack and buffer material. “The Specified Radioactive Waste Final Disposal Act” (The HLW Act) legislated in June 2000 requires the host rock depth deeper than 300ms. Based on the HLW Act, disposal implementer, Nuclear Waste Management Organization of Japan (NUMO) was established in October 2000.

The HLW Act consists of 8 chapters, 10 sections and 94 provisions. Chapters are Chap.1 General rule, Chap.2 Basic Policy and the like, Chap.3 Site Selection, Chap.4 Implement of final disposal and the like, Chap.5 Disposal-implementing entity, Chap.6 Fund management entity, Chap.7 Miscellaneous provision and Chap.8 Penalty.

The Act specifies the overall implementation scheme (Figure 1) and defines the roles and responsibilities of the Government, the implementing organization (NUMO), the funding management organization (i.e. RWMC) and owners of power reactors. Under the Act, the Government (i.e. METI: Ministry of Economy, Trade and Industry) is responsible for settling on the basic policy and final disposal plan for 10-year term in every 5 years. NUMO and RWMC are supervised by METI.

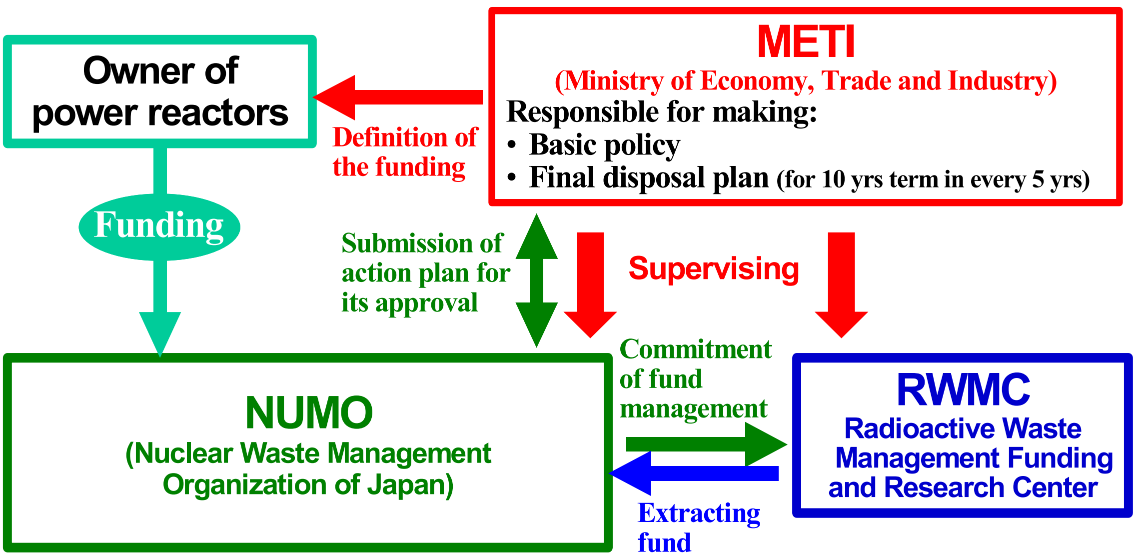

FIG.1. Framework of implementation scheme.

In accordance with the Act, NUMO is responsible to plan and conduct site selection followed by site characterization at the disposal site and relevant licensing application for repository construction/operation and closure. The site selection will be proceeded in a stepwise manner. At the first step, preliminary survey sites will be selected at nation-wide scale mainly by desk survey from the viewpoint of long-term stability of geological environment. At the second step, sites for characterization are then to be selected from the preliminary survey sites where surface based investigations using boreholes will be made to evaluate the characteristics of geological environment for a repository. At the final step, final disposal sites will be selected from the sites for characterization by using an underground facility. A series of these siting processes carried out by NUMO are supervised by the METI. At the every step the NUMO call for opinions to the residents and the METI will call for opinions to the governor and the mayor. The opinions should be respected. As a generator of HLW, owners of nuclear power reactors are responsible to share the cost for final disposal and to provide it to the funding system in accordance with the amounts of electricity generated. The budget for NUMO program should be authorized by METI and assigned from the fund. Management of the funding system is conducted by RWMC, and its activity is also supervised by METI. The total cost is currently estimated at approximately 3 trillion yen (corresponding to 0.13 yen/kWh) for the repository with 40,000 canisters of HLW by the Advisory Committee for Energy, METI.

(c) Waste containing TRU nuclides

The waste containing TRU nuclides generated from RP and MOX fuel fabrication plant have a wide range of radioactivity and wide variety of physical and chemical properties. The Advisory Committee has examined the disposal strategy for the waste and published the report in March 2000. The Advisory Committee concluded that such kind of waste shall be classified according to the above-mentioned properties and shall be disposed of in a manner as NPP waste disposal and geological disposal. In the case of geological disposal, the waste shall be classified into two groups. Waste of the first group are Iodine-129 adsorbent waste and hull and end-peace waste and other waste containing comparatively lower radioactivity are the second group. The first group contains almost all the key nuclides such as Iodine-129 and Carbon-14 and enhanced engineered barrier with bentonite should be applied to delay transportation of radionuclides through engineered barriers.

(d) Uranium waste generated from the uranium conversion and fuel fabrication plant and the uranium enrichment plant

Uranium waste contain only naturally occurring uranium and do not contain other radionuclides that are generated by nuclear activities. Uranium has a very long half-life and it cannot be expected to reduce the radioactivity within a meaningful time period. Moreover radioactivity is increased with time by accumulation of daughter radionuclides. Uranium itself is contained in earth and contributes to natural background. Based on these facts, the Advisory Committee has examined the disposal strategy for the waste and published the report in December 2000. The Advisory Committee concluded that the volume of waste should be reduced by controlling generation of waste at related facilities, and by decontaminating waste to clearance level and reusing recovered uranium. Other waste shall be disposed of in a manner as LLW disposal, however when considering above-mentioned facts and increasing uncertainty of future scenario, model, parameter and data, it should be considered to apply an appropriate dose rate other than 10 μ Sv/y for VLLW and LLW disposal, issued as the exemption dose by the Radiation Council.

(e) Waste generated from radioisotope use, nuclear research and other related facilities

Waste generated from radioisotope use, nuclear research and other related facilities have a wide range of radioactivity and wide variety of physical and chemical properties. A part of these waste also contain harmful materials. The Advisory Committee has examined the disposal strategy for the waste and published the report in May 1998. The Advisory Committee concluded that such kind of waste shall be classified according to the above-mentioned properties and shall be disposed of in a manner as NPP waste disposal and geological disposal. According to the Advisory Committee's estimation, most of waste can be classified into CLW and VLLW. VLLW containing harmful materials should be disposed of in a manner as controlled type disposal facility for non-radioactive waste based on "The Law on Treatment and Clearance of Waste".

1.3. Comparison of basic strategies for waste disposal

Basic strategies for various waste disposal are compared and shown in Table IV.

Table IV. Comparison of basic strategies for radioactive waste disposal

Waste category	HLW	Waste containing TRU nuclides		Uranium waste	Waste generated from NPP		
Waste classification	Geological disposal	Geological disposal	LLW containing comparatively high radioactivity, LLW, VLLW	LLW containing comparatively high radioactivity, LLW, VLLW	LLW containing comparatively high radioactivity	LLW	VLLW
Waste origin	RP	RP, MOX fuel fabrication plant	RP, MOX fuel fabrication plant	Uranium fuel conversion and fabrication plant, Uranium enrichment plant	NPP	NPP	NPP
Disposal concept	Geological disposal	Geological disposal	See NPP waste	See NPP waste	Rock cavern	Concrete pit	Trench
Depth	Under –300ms	ND	See NPP waste	See NPP waste	–50ms to –100ms	–14ms to –21ms	– A few ms
Engineered barrier	Glass, Overpack, Bentonite	Waste form, Cement, or Cement/ Bentonite	See NPP waste	See NPP waste	Waste form, Cement/ Bentonite	Waste form, Cement/ Bentonite	Waste form
Dose standard	ND	ND	ND	ND but other than 10μSv/y should be applied	ND	Normal events: 10μSv/y, Low probabilistic events: not exceed 10 μSv/y significantly	Normal events: 10μSv/y, Low probabilistic events: not exceed 10 μSv/y significantly
Institutional control and period	ND	ND	ND	ND	ND	300y to 400y	50y
Low probabilistic events	ND	ND	ND	ND	ND	Construction work, Residence	Construction work, Residence
Implementer	NUMO	ND	ND	ND	JNFL	JNFL	JNFL, JAERI
Financing system	Fund	ND	ND	ND	Provision and/or Routine operation cost	Provision and/or Routine operation cost	Provision and/or Routine operation cost

Note: ND means not determined yet.

2. ISSUES CONCERNED TO MAKE THE COMMON FRAMEWORK

In Japan, the basic strategies for safe disposal of radioactive waste generated from nuclear facilities have been examined separately. Examination of the common framework has been just started in this year.

2.1. Uranium waste

The safety framework for uranium waste should be separated from the safety framework common to the other radioactive waste from the point of view of rational disposal and natural back ground level. Will it be possible to make an international consensus about this problem?

2.2. Dose standard

10 μSv/y issued as the exemption dose was applied as the dose standard for the NPP waste disposal facility in Japan. Applying such exemption dose standard to the repository after the control period makes it very easy for the people to understand the safety of the repository. However there must be concern that such a low dose standard will cause disposal cost higher than other country's one. For example, a waste that can be disposed of as VLLW in other country shall be disposed of as LLW in Japan, or a design of engineered barriers for LLW disposal facility would become too much robust. Moreover, safety analysis of geological disposal would be more severe when considering perturbation scenario under such low dose standard. The introduction of the intervention level recommended in ICRP Pub.81 is suggested as an approach to this problem. How should regulator or an implementer in other country explain to people about the safety of a disposal facility after the control period without applying the exemption dose level? An international consensus needed on suitable standards for this issue.

2.3. Disposal depth

A host rock for the geological disposal is required in the depth where probability of human intrusion is very low and the perturbation of the ground doesn't cause any effect on the underground facility. The long-term stability of the host rock is also required. It is judged that there will be few human intrusions even in the depth of rock cavern disposal. How should the depth of each facility be determined?

2.4. Reversibility/retrievability

According to "The Basic Policy of Safety Regulations for the Disposal of High-Level Radioactive Waste (The First-Stage Report)" examined by NSC in November 2000, it is said that safety confirmation in each of the phases of construction, operation and other phases is important. It is also said that at the time of the closure of a disposal facility, the results of the safety assessment should be verified under the support of the data obtained in the construction and operation phases and it is important to have a possibility to retrieve the high-level radioactive waste until the appropriateness at the safety assessment results has been confirmed. When considering the common safety framework, the purpose of reversibility/retrievability and its application to a disposal facility should be considered according to waste type.

2.5. Human intrusion scenario

Though depth will have the effect to reduce the human intrusion probability, it is said that predicting such probability as future human behavior is impossible scientifically. The concepts of geological disposal can be categorized into three types, a geological disposal in a rock salt, a geological disposal in an unsaturated rock formation and a geological disposal in a saturated rock formation with multi engineered barriers. The probability and the purpose of human intrusions will be different in each case and the consequence caused by human intrusion will be also different. Will a standard human intrusion scenario be able to be recommended as an international consensus?

2.6. Time frame and safety indicator

It is said that the long-term stability or the long-term behavior of geological condition in Japan can be predicted as far as a hundred thousands of years from now based on the study of the H12 Report. However the human behavior in the future cannot be predicted so far and the peak dose rate will be occurred a million years after disposal or so. If we consider these technical bases, how will it be possible to assess the future doses due to geological disposal scientifically? To make for public confidence building, an international consensus about the time frame and the safety indicator of the safety assessment should be established as far as possible.

2.7. Passive institutional control

The passive institutional control such as a record keeping and limiting land use is considered to be very useful for a rational uranium waste disposal in Japan. It will also be very useful to reduce the concern of the people about a human intrusion in the future. The principle of the safe disposal of radioactive waste is considered not to depend on any institutional control eternally from a technical point of view. However, if such passive institutional control is considered to be effective for a rational disposal and for reducing a concern of the people about a future human intrusion, such a passive institutional control measure should be recommended internationally.

THE BOREHOLE DISPOSAL OF SPENT SOURCES (BOSS)

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Abstract. During the International Atomic Energy Agency (IAEA) Regional Training Course on "The Management of Low-Level Radioactive Waste from Hospitals and Other Nuclear Applications" hosted by the Atomic Energy Corporation of SA Ltd. (AEC), now NECSA, during July/August 1995, the African delegates reviewed their national radioactive waste programmes. Among the issues raised, which are common to most African countries, were the lack of adequate storage facilities, lack of disposal solutions and a lack of equipment to implement widely used disposal concepts to dispose of their spent sources. As a result of this meeting, a Technical Co-operation (TC) project was launched to look at the technical feasibility and economic viability of such a concept. Phase I and II of the project have been completed and the results can be seen in three reports produced by NECSA [1 – 3]. The Safety Assessment methodology used in the evaluation of the concept was that developed during the ISAM programme [4] and detailed in Van Blerk's PhD thesis [5]. This methodology is specifically developed for shallow land repositories, but was used in this case as the borehole need not be more than 100m deep and could fit into the definition of a shallow land disposal system [6]. The studies found that the BOSS concept would be suitable for implementation in African countries as the borehole has a large capacity for sources and it is possible that an entire country's disused sources can be placed in a single borehole. The costs are a lot lower than for a shallow land trench, and the concept was evaluated using radium (226) sources as the most limiting inventory. The conclusion of the initial safety assessment was that the BOSS concept is robust, and provides a viable alternative for the disposal of radium needles. The concept is expected to provide good assurance of safety at real sites. The extension of the safety assessment to other types of spent sources is expected to be relatively straightforward. Disposal of radium needles is believed to be a more severe test of the disposal concept than any other type of spent source. Phase III, which has as yet not started will, look at developing guidelines for implementation of the concept and will end with a practical demonstration of the concept.

1. INTRODUCTION

With a view to improving radioactive waste management on the African continent, the NECSA started to identify and review disposal concepts that could be applicable to African conditions. A few factors that were taken into consideration include:

- A nuclear infrastructure that can control a sophisticated disposal facility, is basically nonexistent in African countries;
- The disposal concept must be technically feasible to implement, taking into consideration the equipment at their disposal;
- Relative to conventional methods, the concept should be easy to manage;
- The concept must be economically feasible to implement, taking into consideration the financial situation in some of these countries;
- The concept must take into consideration the size and number of sources (i.e. the volume) that need to be disposed of;
- The concept should comply with the overall safety objective of radioactive waste management, in terms of the protection of human health and the environment both now and in the future, without imposing undue burdens on future generations; and
- The concept should preferably be applicable for both short- and longer-lived radionuclides, such as ^{226}Ra and ^{241}Am.

This led to the introduction of the borehole disposal concept as a near surface disposal method of spent radioactive sources in African countries. As part of an IAEA programme to strengthen waste management infrastructure in African countries (AFRA I–14), an IAEA TC Project was awarded to NECSA to investigate the borehole disposal concept.

1.1. Objective of the project

The objective of the project was to validate the technical feasibility and economic viability of a "borehole disposal" facility to be used for the safe disposal of Disused Sealed Radioactive Sources (DSRS).

1.2. Terms of reference

The purpose of the study was to demonstrate the possibilities and limitations of the borehole disposal concept. Emphasis was placed on the following:

— Defining the required safety criteria.
— Defining and justifying a complete and coherent set of assessment scenarios, including an intrusion scenario.
— Assessment of compliance with safety requirements.
— Deriving generic waste acceptance criteria for spent sources to be disposed of in the borehole.
— Assessment of the technical and economic feasibility of the borehole concept.
— Definition of a representative waste description for Africa.
— Investigating the possibilities, limitations and requirements of a mobile conditioning facility (including the necessary working procedures).
— Designing and implementation of a pilot plant, including the waste packages decided on.
— Designing of the waste packages, justifying the use of materials and dimensions.
— Investigating the technical and economic feasibility of alternative borehole diameters.

In the final report, reference will specifically be made to the following aspects:

— Minimum site characterisation necessary.
— Institutional control.
— Depth requirement for the disposal of different isotopes (sources).
— The use of sensitivity analysis in the safety assessment.
— Sorption properties of different isotopes on materials.
— The influence of geochemistry on the waste package design.
— Drilling methods that should be used.
— Public acceptance.

Note that the safety assessments, as part of the evaluation of the long-term performance of the disposal concept, were conducted within the framework established in the IAEA Co-ordinated Research Programme on *Improving Safety Assessment Methodologies for Near Surface Radioactive Waste Disposal Facilities* (ISAM) [4].

1.3. Acronym for the concept

Several studies have been, or are currently under-way to investigate the feasibility of boreholes as an appropriate disposal facility. It may be that these investigations consider wider and deeper or shallower boreholes, and for different inventories. To eliminate any confusion regarding other concepts that use boreholes, it is proposed that the concept presented in this document be referred to as the BOSS disposal concept (Borehole Disposal Of Spent Sources).

2. THE PROBLEM WITH DISUSED SEALED RADIOACTIVE SOURCES (DSRS)

Large numbers of spent radiation sources exist in many countries. At last count [7], there were more than 600 000 sources in existence world-wide. A summary of international experience in using boreholes to store/dispose of sources can be seen in an annexure to a draft report currently being developed at the IAEA [8], but as yet, no international consensus exists as to an acceptable solution for the challenge of disposal of Disused Sealed Radioactive Sources (DSRS). Controlling of these small and active sources is particularly urgent in developing countries that usually do not have the necessary infrastructure to deal with the problem. Some countries in Africa do not even have legislation for the control of radioactive waste, let alone the dedicated staff to collect and store these sources.

The Field Management Committee (FMC) of the African Regional Co-operative Agreement (AFRA) has taken steps to rectify the problem with respect to radium needles and to date eight African countries have been "cleared" of known radium sources. These sources have been conditioned and placed in safe storage in the respective countries and will be stored until a disposal route exists. Part of the TC project [2] included obtaining detailed inventories of sources from various African countries.

3. THE CONCEPT [2]

The design of the borehole for the BOSS disposal concept (see Figure 1) should contribute to the general aim of safety for near surface disposal systems, that is, to ensure the safety of the public and the environment now and in the future. Fundamental aspects to take into consideration in the design of the borehole are:

— The dimensions of the borehole should allow for the disposal of spent sources in suitable waste packages;
— The design of the borehole should take into consideration the operational requirements, e.g. waste emplacements should be able to take place as a matter of routine over the period during which it operates;
— The design should minimise the need for active maintenance after site closure and complement the natural characteristics of the site to reduce environmental impact; and
— Human intrusion (advertent and inadvertent) should be difficult.

Conceptually, the BOSS disposal concept comprises a standard borehole drilled down to a depth of 100 m. The standard borehole (165 mm or 6.5" in diameter) and the 100 m are guidelines. If required, wider boreholes can be drilled. Also, for site-specific conditions, depths of less or more than 100 m could be acceptable.

A 150 mm casing is used to define the disposal volume. To ensure that the disposal volume is dry during the operational period, a bottom plug is provided. The disposal area can be fenced off to limit access, and a temporary site office can be erected.

A reference design was proposed in the studies undertaken by Van Blerk, et al. [2] that included a container of stainless steel, a cement based waste form and encapsulated sources. The waste package would be placed into wet concrete in the borehole. A specially formulated concrete would then be poured on and around the container. The next package would then be lowered into the hole and the process repeated. Packages would continue to be placed into the borehole until the Waste Acceptance Criteria for that hole were met or until the cut-off depth was reached. The rest of the hole would be sealed off with concrete.

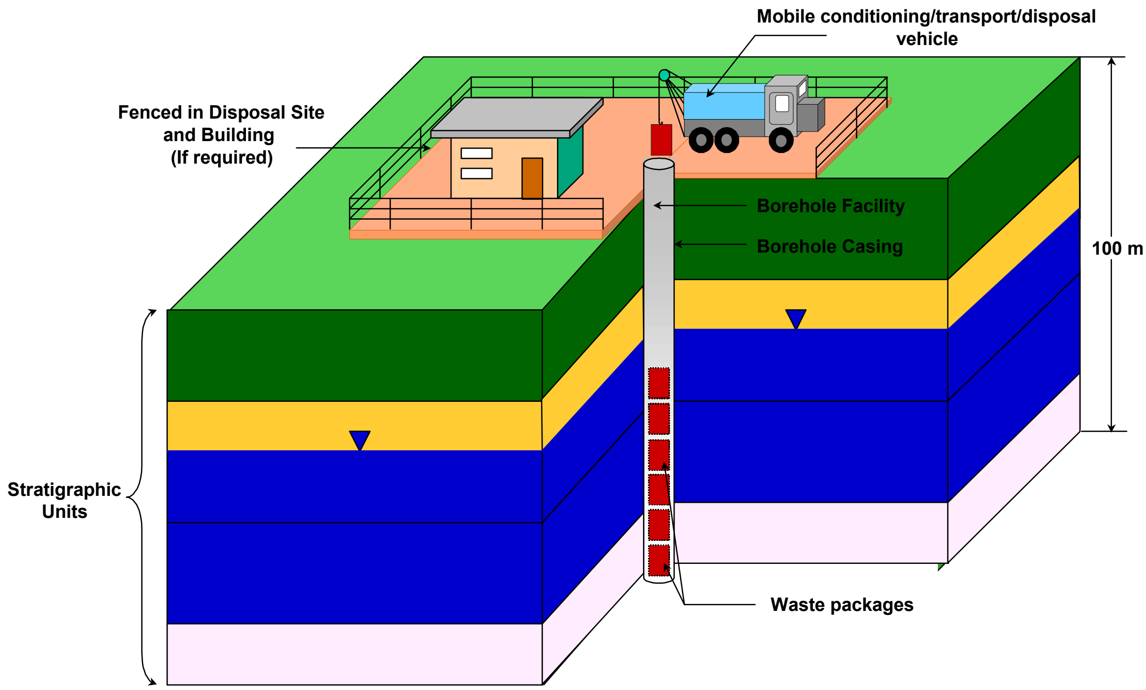

FIG. 1. A schematic representation of the BOSS concept (after [2]).

Whether the site should be marked or not is an interesting philosophical point to debate, as the “footprint” of a borehole is very small and there is merit in possibly putting some sort of intrusion resistant cap at a shallow depth and then camouflaging the hole so that it cannot be found. This issue can only be resolved by a detailed Safety Assessment.

Retrievability has not been considered for disposal in boreholes as the concept has been developed to solve specific problems in developing countries, especially Africa. It is currently felt that building in this factor would unnecessarily complicate the technical requirements and make the implementation impossible in the target countries.

The preliminary studies of the economics of the BOSS concept have shown that the costs per cubic meter of waste are higher than for some shallow land sites, but this is because the small volumes carry a large “overhead” in terms of siting, radiation protection etc. The total cost of implementing the concept in a country could range from US $123 000 to US $213 000 [2]. The total capital cost would of course be a lot less than that of conventional disposal concepts.

4. THE PRELIMINARY SAFETY ASSESSMENT [3]

The Safety Assessment was carried out using real data from two sites, namely the Vaalputs Shallow Land Repository site, and the Pelindaba site, both of which are in South Africa. The Vaalputs site used a shallow borehole of 45m below surface in the unsaturated zone, while the Pelindaba hole was 100 m deep in the saturated zone.

Three scenarios were considered for each site, namely:

- Design Basis Scenario (Drilling a well for water near the repository).
- Drilling Scenario (Direct Intrusion).
- Modified Society Scenario (Termites at Vaalputs, discharge at spring at Pelindaba).

The Design Basis Scenarios for Vaalputs (V1) and Pelindaba (P1) reflected similar behaviour patterns to ones currently practised at the sites and exposures were assumed to come from a well drilled near the waste. Direct intrusion (Scenarios V2 and P2) by means of a well drilled directly into the waste was considered highly unlikely and was not modeled in the assessment. The modified society scenario for Vaalputs (V1) considered alternative lifestyle assumptions. The hunter/gatherer was assumed to make up the critical group and the only pathway was via the activities of termites. This scenario (modified society) at Pelindaba (P1) considered the discharge of groundwater through a spring at surface.

The BOSS disposal concept easily met generally acceptable safety criteria for almost all analyses. One exception is the analysis of direct consumption of contaminated termites. As this scenario is based on very limited knowledge of the behavior of termites, very conservative assumptions were made. The assessment [3] assumed a high loading of radium needles in a waste package and a high loading of waste packages in a borehole. Despite this level of loading, the disposal facilities demonstrated acceptable levels of safety for most conditions considered. It is concluded from this evaluation [3] that the waste package need not be limited in inventory. A good waste form is critical for meeting the safety criteria and therefore good quality control would be needed whilst performing the waste conditioning and emplacement.

5. CONCLUSION

From a financial perspective, the BOSS concept is cheaper than a shallow land concept even though one would not consider placing radium sources, in particular, in such a repository. It does not make sense to compare the BOSS concept financially with conventional disposal systems that are designed to dispose of very much larger volumes of waste. Of course, if deep geological facilities already existed, the problem of DSRS would be easier to solve. The BOSS design appears to be quite robust in terms of limiting releases to acceptable levels. This was found to be true for disposal in both saturated and unsaturated conditions, although unsaturated conditions appear to be preferred [3]. From the work done in the TC project, [2, 3] it can be concluded that that the BOSS concept is robust, and provides a viable concept for the disposal of spent radium needles. Based on the results obtained with the two generically defined sites, the concept is expected to provide good assurance of safety at real sites. The extension of the safety assessment to other types of disused sources is expected to be relatively straightforward. Disposal of radium needles is believed to be a more severe test of the disposal concept than any other, and since a strong safety case can be compiled for ^{226}Ra, one would expect that the concept will prove to be safe for any other type of disused source.

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A.3. MAKING THE SAFETY CASE – DEMONSTRATING COMPLIANCE

SAFETY CASE: AN INTERNATIONAL PERSPECTIVE

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Abstract. In recent years, it has become more and more evident that repository development will involve a number of stages punctuated by interdependent decisions on whether and how to move to the next stage. These decisions require a clear and traceable presentation of technical arguments that will help in giving confidence in the feasibility and safety of the proposed concept. The depth of understanding and technical information available to support decisions will vary from step to step. A safety case is a key item to support the decision to move to the next stage in repository development. Progress is noted, in the past decade, in the performance and safety assessment areas, particularly in the methodologies for repository system analysis. Progress is also observed regarding the understanding of the natural system and its characterisation, treatment of uncertainties, and modelling. Some areas are under active development, e.g. the area of scenario development and analysis. Finally, to increase confidence, rigorous quality assurance procedures need to be implemented, as well as the factoring of the contribution of R&D in underground research laboratories. The paper summarises the lessons learnt within relevant NEA initiatives as they evolved over the course of a decade and now allow a comprehensive view of what constitutes a safety case.

1. INTRODUCTION

Implementing and regulatory organisations in many of the OECD/NEA member countries are involved in the investigation and resolution of issues associated with long-term safety of underground repositories for radioactive waste.

In every national radioactive waste management programme, it is recognized that an appropriate level of confidence in the achievability of long-term safety should accompany the decision at each stage of a step-wise repository development. The current level of technical confidence should be illustrated and argued in a safety case. A safety case should also argue the possibility of reducing the current level of uncertainty in the next development phases.

The concept of a "safety case" has been progressively clarified in a series of initiatives that the NEA has had in the past decade, and which culminated with the publication of the NEA confidence document [NEA 1999] and the latest IPAG exercise [NEA 2001].

> "*Safety Case: A Safety Case is a collection of arguments at a given stage of repository development, in support of the long-term safety of the repository. A Safety Case comprises the findings of a safety assessment and a statement of confidence in these findings. It should acknowledge the existence of any unresolved issues and provide guidance for work to resolve these issues in future development stages*."

> *"Safety Assessment: is the evaluation of long-term performance of compliance with acceptance guidelines and of confidence in the Safety indicators by the assessment results."*

> *"Performance Assessment: the analysis of the performance of the system concept, with the aim of developing confidence that the system will (or can be designed to) perform within acceptable bounds."*

Over the years, insights were obtained from the activities of the working group IPAG, from reviews of national safety studies, from the confidence document, topical sessions within the past PAAG and SEDE groups, plus from the cumulative knowledge of the IGSC. The findings of the various relevant initiatives are reviewed hereafter.

2. FEEDBACK FROM THE THREE IPAG PHASES

The IPAG initiative started in 1994 with the aim to provide an international platform to examine the overall status of Safety Cases and their supporting Integrated Performance Assessment (IPA) studies. The work has been carried out in three phases. IPAG-1 from 1995 to 1996, aimed to examine completed IPA studies as a practical body of evidence that would indicate the current status[1] of PA. IPAG-2 from 1997 to 1998 was carried out in order to examine the experience of peer reviews of IPAs, and especially reviews performed in support of regulatory assessment, from both the implementer and regulator points of view[2]. IPAG-3 has been carried out mainly between June 1999 and November 2000. It focused on the evaluation of the state-of-the-art for obtaining, presenting and demonstrating confidence in long-term safety and, made recommendations on future directions and initiatives for improving confidence[3].

The increase in the number of national organisations (e.g. from ten during IPAG-1 to 20 in IPAG-3) participating in these exercises stresses the importance of the Safety Case. The findings, such as compilations of questions and answers and the achievements of discussions during the successive IPAG exercises, constitute a relevant database of national Safety Cases.

2.1. IPAG 1: "Lessons learnt from ten performance assessment studies"

On the basis of the examination of ten PA studies, the IPAG-1 group identified several areas in which significant advances in methods and applications, and prospects for specific improvements were observed.

Traceability[4] and *Transparency*[5] were considered as relevant issues in PA even though they consume time and resources. Some guidance on methods for promoting transparency is given such as the presentation of assumptions and their basis, modelling accurately, data used and their sources, explanation of the results and points of weakness.

According to the Safety Assessment report, participants agreed that a universal plan of contents couldn't be recommended. However, as said before and taking account of the need to promote transparency in PA documents, a set of fourteen elements (or topics) that should be addressed in a safety assessment report was proposed:

[1] IPAG-1: Lessons Learnt from Ten Performance Assessment Studies, OECD/NEA 1997 report [NEA, 1997].
[2] IPAG-2: Regulatory Reviews of Assessments of Deep Geological Repositories, OECD/NEA 2000 report [NEA, 2000-a].
[3] IPAG-3**:** Approaches and arguments to establish and communicate confidence in safety and the overall results of IPAs, OECD/NEA report (to be published).
[4] By traceability, it was understood as an unambiguous and complete record of the decisions and assumptions made, and of the models and data used in arriving at a given set of results.
[5] By transparency, it was understood as the PA record to be written in such a way that its readers can gain a clear picture, of what has been done, what the results are and why they are as they are.

Program context	Historical perspective, regulatory context, brief description of the waste disposal concept
Regulatory criteria	Criteria or guidance, quantitative and qualitative
Objectives and scope of the assessment	Related to the programme context
Description of the system at the conceptual level	Required level of safety e.g. multi barrier concept, safety functions
Statement of the constraints	Long time scale, uncertainties.
Approach to safety assessment	Treatment of uncertainties, models using, traceability…
Detailed description of the disposal system	Waste form, EBS, site characteristics
Interpretation and elicitation of databases	Methods description, use of expert to elicit data
Scenario development	Methodology, description, assumptions, justification
Description of models	Conceptual and mathematical, spatial and temporal, assumptions
Results and interpretation	For individual subsystems and total system, sensitivity uncertainties.
Confidence in key arguments	Key processes, model, data and assumptions revisited and their basis examined
Compliance with regulatory criteria	Overall compliance with regulatory criteria
Conclusions	Indication of areas in which further development is required, which goals have been reached

IPAG 1 participants also revealed the different vocabularies to be clarified such as Performance Assessment/Analysis, Safety Assessment/Analysis/Case and agreed there was no available definition, but an interpretation might be given within the safety study.

IPAG-1 participants pointed out the potential long-term stability uncertainties as far as the geosphere as a key component of the multi-barrier system is concerned. They recommended coordination between site characterisation and design such as an explanation of the basis for selecting geosphere functions to be taken into account and to be discarded in PA calculations.

Moreover, PAs should address the issues of uncertainties (on scenarios, models and parameters) and completeness in the context of the safety arguments and relevant characteristics of the specific disposal system. Finally, natural analogues were seen as a component of the confidence-building process as they support the understanding of key processes regarding the different components of the multibarrier system and provide evidence that no unexpected processes or phenomena have been present or active.

2.2. IPAG 2: Regulatory reviews of assessments of deep repositories

The IPAG-2 study compared international experiences of peer reviews of IPAs and, especially reviews performed in support of regulatory assessments from both implementers and regulators points of view. The findings on four topics are summarized hereafter:

As far as the conduct of review is concerned, it was observed that the dialogue, and moreover making records and stable documentation from this dialogue, is important and of benefit both for regulators and implementers and enhances the overall credibility of the step-wise process through which implementer and regulator should establish a structured framework for PAs and reviews early in a repository programme. The implementer has to produce a complete analysis of repository safety; thus independent assessments or calculations by regulators can be beneficial for the reviews. Regulatory guidance should clearly state the requirements and be flexible in order to avoid frequent updates.

Concerning the Safety Case, the necessary integration of site investigation and design development in the PAs was pointed up, focusing on the intrinsic safety of the disposal system and not only on the calculations used to demonstrate safety. Multiple lines of reasoning such as natural analogues and paleohydrogeology might help in this objective and variety of techniques and approaches could be used in a complementary manner, e.g., probabilistic versus deterministic calculations. In addition, qualitative "soft" and quantitative "hard" information should be considered as complementary arguments and as already indicated in the IPAG-1 report; consideration should be given to increasing their value in the decision-making process. At the end, the multi-barrier system was confirmed as the key element for long-term safety and further work is needed to describe what is specific to a deep geological disposal system. Implementers clearly need also to be explicit in their interpretation of the multi-barrier concept.

In support of the above thesis on the traceability and transparency stressed in the IPAG 1 study, IPAG 2 also advised that IPAs prepared for licensing purposes be traceable, transparent, reproducible and publicly available. One aspect of developing traceability and understanding between the implementer and regulator is consistency of the methods and documentation structure and style. Other, non-technical stakeholders also review IPAs and have different needs with regard to traceability and transparency. An integration of their viewpoints is needed. At the end, NEA should explore in greater detail the approaches and techniques used for addressing the needs of public and other non-technical stakeholders in IPAs.

2.3. IPAG 3: Approaches and arguments to establish and communicate confidence in safety and the overall results of IPA's

In IPAG-3, the objectives were to evaluate the state-of-the-art for obtaining, presenting and demonstrating confidence in long-term safety, and make recommendations on future directions and initiatives for improving confidence. The IPAG-3 study mainly followed the work done on confidence building in 1999 in which the terms "Safety Case" and "IPA" are distinguished. The IPAG-3 study continued to make a distinction between these two terms. During IPAG-3 discussions, three main topics were developed for which some messages are summarized hereafter.

As already mentioned in the previous IPAG exercises, the long duration of the process reflects the complexity of the tasks and the desire to proceed by cautious steps due to technical issues and social acceptance.

These decisions may regard, for example, interim surface storage, siting and design, safety assessment, site characterisation, and the licensing of construction, operation and closure, sealing and post-closure monitoring that are taken throughout the development of a facility. Generally, the Safety Case is considered as one of the key bases and needs thus to be structured, technically argued, and supported with a clear link to the step-wise decision-making process and the level of confidence must reflect commitments at each relevant step. Based on answers provided by IPAG-3, key arguments were identified and categorized as presented below:

Confidence in the Proposed Disposal System	— Intrinsic robustness of the multi-barrier system — "What if?" scenarios and calculations — Comparisons with familiar examples and natural analogues
Confidence in the Data and Knowledge of the Disposal System	— Quality of the research programme and site investigations — Quality assurance procedures — Data from a variety of sources and methods of acquisition — Use of formal data-tracking techniques
Confidence in the Assessment Approach	— Logical, clear, systematic assessment approach — Assessment conducted within an auditable framework — Building understanding through an iterative approach — Independent peer review of approach
Confidence in the IPA Models	— Explaining why results are intuitive — Consideration of alternative conceptual models and modelling approaches – simple and complex — Testing of models against experiments and observations of nature — Model comparison exercises — Comparisons with natural analogues — Independent evidence such as paleohydrogeological information
Confidence in the Safety Case and the IPA Analyses	— Clear statements and justifications of assumptions — Demonstrate that assumptions are representative or conservative — Sensitivity studies — Clear strategy for managing and handling uncertainty — Multiple safety indicators — Multiple lines of reasoning
Confidence via Feedback to Design and Site Characterisation	— Support for any disposal concept design changes — Overall quality and safety of the disposal system

Uncertainty regarding phenomena and data over time scales such as the future course of events external to the repository or the long-term evolution of engineered materials were emphasized as an important issue in the Safety Case. Consequently, IPAG-3 raised the need for a clear strategy for dealing with these uncertainties that will be explained within the Safety Case and its supporting IPA. Generally, the higher the margins of safety in barrier performance, the less stringent are the demands on the precision of associated data. The quality of the Research and Development (e.g., site characterisation, properties of materials involved…), procedures, data and the use of these data are crucial and must be clearly stated in a dedicated section of the IPA's documentation.

Moreover, IPAG-3 participants confirmed that the assessment and modelling approaches must also be clear, logical, continually improved through an iterative process, and submitted to independent peer reviews and accepted by stakeholders. For getting confidence in the modelling, one solution might consist of verification of numerical modelling by comparison with simple analytical models, solution of test problems and comparison of results from different methods used to resolve the same problems. Another solution involves participating in international model comparison exercises, using models to predict measurable parameters, comparison of results with natural analogues, and using paleohydrogeological data.

IPAG-3 participants recommended that a Safety Case should include a clearly developed "confidence statement", as proposed in NEA [1999], and that this should be given a prominent location within the documentation. The confidence statement should explain clearly how the assessment results compare with the appropriate regulatory criteria, and could also make comparisons with levels of naturally occurring radiation and other everyday risks, to put the radiological risks arising from a repository into perspective.

Regarding the communication and presentation part of IPAG-3, the Safety Case is viewed as a starting point for related presentations, brochures, etc. to different stakeholders and for which their expectations must be understood. The presentation of a Safety Case must not underestimate the audience's technical sophistication so that the scientific community can be a participant in the societal decision-making process. Communicating a Safety Case will require multiple presentations because of the complexity of the integration process and the multitude of scientific and engineered disciplines. Flexibility needs to be maintained in consistency with audience's requests. Delivered messages must be clear and one 'main message' for a given iteration of a Safety Case should be recommended.

As a conclusion, the Safety Case is a management issue. Feedback to each relevant part of the Safety Case such as design and site characterisation can be possible if it is planned and managed at each relevant step of the repository development. As regards this last point, it is suggested that it is perhaps an area where further work could be done.

3. FEEDBACK FROM PEER REVIEWS AND CONFIDENCE DOCUMENT

Peer reviews are considered as a relevant activity of the RWMC division in the NEA. The peer reviews are both on methodologies and major R&D studies and to provide a decision-making basis for moving from one step to another. Peer Reviewers want to see more of a Safety Case than a Safety or Performance Assessment. Vis-à-vis a SA/PA, a Safety Case places the analyses and information in a decision-making context and shows strategic thinking and management strategy. They generally mention that the programme needs to be performed in a way that favours dialogue with stakeholders: technical community, regulators, and local community.

The previous peer reviews generally show that:

- Safety Assessments are just built around assessment capability and "regulatory requirement".
- The geology is normally well described, but more data are needed.
- Engineered barrier systems need to be emphasised in that they offer, at early stages, margins for optimisation and adaptation to the local environment.
- The role and use of experts is an important issue in building confidence.
- Less technical sections do exist but are not accessible to non-specialists. Lots of details exist with little rationale for choices/decisions, e.g., role of scenarios, role of biosphere representations, data.
- Lessons are being generated for subsequent assessments or repository development phases, but this feedback is not evident.
- Reasons for confidence in the analyses are interspersed in the documentation. No drawing together in conclusions on the quality of what was done and perspective on what more may be done.
- Common finding is thus that these studies have important limitations as tools for:

  - informing decisions on what to do next
  - dialogue amongst interested parties
  - arguing safety.

Typical problem areas are identified such as:

- Completeness of scenario analysis
- Consistency needs to be improved in assumptions, the level of detail, the vocabulary
- The transitory phase needs more attention
- Issues of traceability and transparency.

The NEA report on "Confidence in the Long-term Safety of Deep Geological Repositories" was published in 1999. It noted that a viable repository project depends on confidence in safety on the part not only of technical specialists in implementing and regulatory organisations and in the wider scientific community but also of political decision-makers and the general public. Various topics as described in the previous paragraphs are emphasised in this document, such as the step-wise process of deep geological repository development and the necessity to preserve credibility and confidence during the process. It is also mentioned that the Safety Case for a repository is carried out firstly by the implementer in a framework set by a number of practical and programme specific constraints. The articulation and contents of four main processes (see figure hereafter) that need to be achieved will permit safety to be argued and assessed by decision makers such as regulators, scientific bodies, and/or others:

(i) Establish an ASSESSMENT BASIS

– define a safety strategy that describes a suitable approach to the building of a safety case
– define the repository site and design (*system concept*)
– define the *assessment capability*

(ii) Carry out a PERFORMANCE ASSESSMENT

– evaluate repository performance for the assessment cases
– assess compliance with acceptance guidelines
– carry out sensitivity analyses

(iii) EVALUATE CONFIDENCE in the calculated safety and modify, if necessary the assessment basis

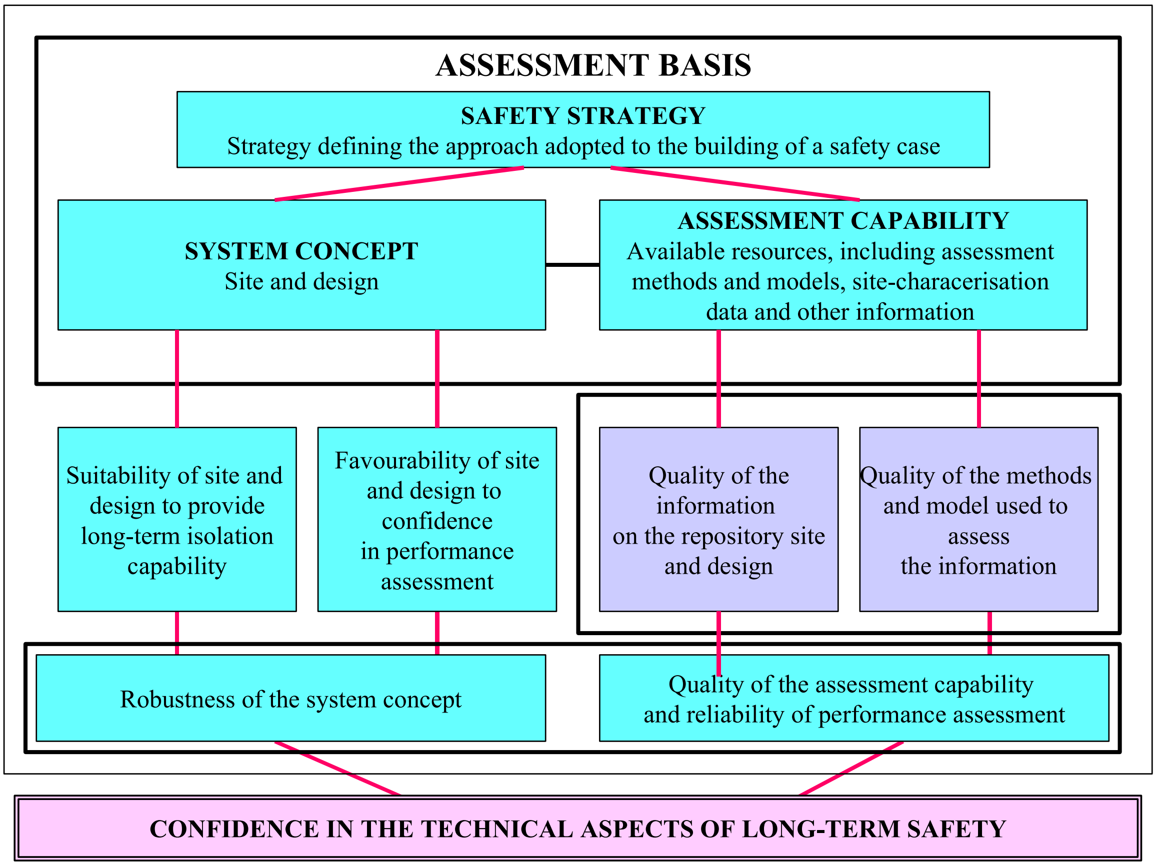
Steps (i), (ii) and (iii) define the safety assessment.

(iv) Compile a SAFETY CASE

– document the safety assessment
– state confidence in the safety indicated by the assessment and describe proposed way forward

Interact with decision makers and modify, if necessary, the assessment basis

The assessment basis is in fact the combination of the safety strategy, the system concept, and the assessment capability as described in the figure below:



To achieve safety, the confidence document raises the need to provide a conscious "safety first" approach that could be attempted regarding various possibilities such as:

— Through site and design choices, avoiding or forcing to low probability or consequences most phenomena that could be detrimental to safety
— A further characterisation and means to reduce uncertainty through R&D,
— Continuity of means and resources
— Avoiding over-reliance on any single safety provision
— Identified process for acquisition of technical knowledge and tools
— Internal guidelines showing a controlled, fit-for-purpose programme
— Periodic programme and quality reviews
— QA procedures to minimise likelihood of defects and errors
— Openness towards dealing with varied technical opinions (inside and outside programme).

Achieving safety is not sufficient to get confidence in the safety case. One key task is to provide relevant arguments on the confidence in the safety. These arguments must be declared and described in the Safety Case A confidence statement should describe the strength of the arguments on which the findings of the safety assessment are based. For instance, in order to better support the confidence statement, it is important to offer commentaries on (i) the role of barriers and system functions, (ii) the identification and explanation of the assessment cases, (iii) the verification of the quality of tools, data, analyses, and the explanation that PA is for testing system performance.

4. FEEDBACK FROM IGSC EXCHANGES

From a variety of feedbacks, the Integration Group for the Safety Case [*IGSC*] considers that the Safety Case is one of the main work objectives of national programmes, which provides the rationale for doing work within the IGSC. Using the NEA confidence document as a starting point, the IGSC will continue work on the definition of process, components, methodology and means of ensuing consistency which are required to build a Safety Case. The IGSC has a role to develop common views on such key aspects of the Safety Case but should not be prescriptive. Since the beginning of the IGSC, two meetings were organised with topical sessions. The following major messages emerge from these exchanges:

- Multiple lines of reasoning should include additional safety measures and indicators.
- It is not possible to rigorously demonstrate compliance; the only realistic objective is to achieve adequate confidence.
- The way in which different bodies of scientific opinion are dealt with in the Safety Case is an important and outstanding issue.
- Whether, for example, operational safety is included in the Safety Case will depend on the particular circumstances of the Member countries.
- Development of the Safety Case involves mediation with society.
- We should take a common sense definition of the Safety Case and not make it more complicated than it needs to be.
- It is a presentation and linking of information and arguments on safety needed to support the decision-making process.
- Dependent on the programme-specific and regulatory context, the implications of retrievability may need to be dealt with in the safety case.

Different countries are at different stages and therefore opinions can be expected to vary on where the key issues remain.

5. CONCLUSION AND FUTURE WORK

The successive IPAG exercises have shown a clear evolution from a calculation (Performance Assessments) to integration (Safety Case) approach. The Safety Case is more than calculation results. It includes both qualitative and quantitative arguments and is not a science-only product.

At a technical level, the most important issue is how to manage dialogue with technical experts both in-house and outside. Communication with both technical and non-technical stakeholders is a priority goal for building confidence. A Safety Case should provide a well-documented safety assessment, including an evaluation of confidence in the calculated safety. It should describe the approaches adopted to achieve confidence and should include a formal *statement of confidence*. This last point corresponds to the fact that all relevant data and information, and their uncertainty, have been given consideration, all models have been tested adequately and a rational assessment procedure has been followed. Additionally, results need to be fully disclosed and subjected to QA and review procedures. The Safety Case is about managing and integrating technical and non-technical information. “Management” – “Safety culture” – “Strategy” – “Confidence” are key words in confidence building and for preparing a credible Safety Case.

The NEA Confidence Document and the IPAG-3 document are the culmination of a decade of efforts at the NEA in the area of development of the safety case. The IGSC is now the working party in charge expressly of developing all Safety-Case-related technical activities.

Aspects related to non-technical stakeholders will also be taken into account by the IGSC, but will also be reviewed by the NEA Forum on Stakeholder Confidence.

Consistent with the different initiatives carried out since 1990, one of the activities that will start under the NEA IGSC platform will concern the preparation of a booklet on the Safety Case. It will describe the issues connected to the Safety Case and the approaches available for satisfying the four elements of a Safety Case. It will not imply that all techniques and approaches be used in every safety case. Rather, it would provide a sense of the problems that exist and the range of techniques and approaches that can be used to formulate the Safety Case and develop and communicate confidence.

PERSPECTIVE ON SAFETY CASE TO SUPPORT A POSSIBLE SITE RECOMMENDATION DECISION

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Abstract. The mission of the US Department of Energy (DOE) is to provide the basis for a national decision regarding the development of a geological repository for spent nuclear fuel and high-level radioactive waste at the Yucca Mountain site in Nevada. There are a number of steps in the decision process defined by US law that must be completed prior to development of a repository at this site. The DOE's focus is currently on the first two steps in this process: characterization of the site to support a determination by the DOE on whether the site is suitable for a geologic repository and a decision by the Secretary of Energy (the Secretary) on whether to recommend to the President that the site be approved for a repository. To enhance the confidence of multiple audiences in the basis for these actions, and to provide a basis for subsequent action by the President and the US Congress, information supporting the decision process must include the elements of a safety case consistent with the statutory and regulatory framework for these decisions.
The idea of a safety case is to broaden the basis for confidence by decision-makers and the public in conclusions about safety. A safety case should cite multiple lines of evidence, or reasoning, beyond the results of a safety assessment to support the demonstration of safety, which includes compliance with applicable safety criteria. The multiple lines of evidence should show the basis for confidence in safety. To be most effective, such evidence requires information not directly used in the safety assessment.

1. OVERVIEW

1.1. Statutory framework for decisions

The Nuclear Waste Policy Act of 1982 (NWPA) [1] established US policy on waste management and a framework for decisions leading to disposal of high-level radioactive waste and spent nuclear fuel in geologic repositories. The NWPA was amended in 1987 [2] to focus the site characterization process on a single site – the Yucca Mountain site in Nevada – to determine its suitability for recommendation for development as a geologic repository. The NWPA established the responsibilities of several federal agencies in the repository development process. DOE is responsible for repository siting and development activities. The DOE is also required to develop guidelines establishing the criteria that it will use in evaluating the suitability of repository sites. The US Environmental Protection Agency (EPA) is required to develop radiation protection standards for repositories. The US Nuclear Regulatory Commission (NRC), as the licensing authority, is responsible for developing the criteria that it will use in deciding whether to approve construction, operation, and eventual closure of repositories.

The NWPA, as amended, defined the following steps in the national decision process leading to development of a geologic repository:

— The DOE must evaluate the Yucca Mountain site to determine its suitability for development as a geologic repository.
— If the site is suitable, the Secretary of Energy must decide whether to recommend to the President that the President approve the site for development as a repository. An Environmental Impact Statement (EIS) and other prescribed information must accompany a site recommendation.

— If the Secretary recommends approval of the site, and if the President considers the site qualified for application for a construction authorization, the President must submit a recommendation to Congress.
— If the President recommends approval of the site to Congress, the designation of the Yucca Mountain site for a repository must be approved as required by the NWPA.
— If Yucca Mountain is designated as the site for development of a geologic repository, the DOE must obtain NRC approval first for repository construction, then for an operating license, and, eventually, for repository closure and decommissioning.

The Energy Policy Act of 1992 (EnPA) [3] directed a revision of the regulatory framework for the decision process. The EPA was directed to establish site-specific radiation protection standards for a potential Yucca Mountain repository. These standards are to be based on and consistent with the recommendations of the US Academy of Sciences (NAS) and are to be the only standards applicable to a Yucca Mountain repository. The NRC was directed to revise its licensing criteria to be consistent with the EPA's site-specific standards and the NAS recommendations.

1.2. Regulatory framework for decisions

The EPA proposed site-specific radiation protection standards for a potential Yucca Mountain repository in 1999 [4] and issued its final standards at 40 CFR (Code of Federal Regulations) 197 in June 2001 [5]. The final rule establishes standards for both pre-closure and post-closure repository performance. The post-closure standards include all-pathways annual dose standards for individual protection and for a prescribed human intrusion scenario. The rule also establishes a separate standard for protection of ground water. The post-closure standards apply to the accessible environment outside a controlled area, which can extend no more than 18 km from the repository in the predominant direction of ground water flow. The time period for compliance with these standards is 10,000 years. An evaluation of the peak dose, if it occurs after 10,000 years, is to be included in the DOE's EIS.

The NRC is finalizing the licensing criteria for a potential Yucca Mountain repository that it proposed in 1999 at 10 CFR Part 63 [6]. The NRC's licensing criteria proposed at 10 CFR Part 63 would specify the performance objectives for the geologic repository that must be met through permanent closure and those performance objectives that must be met after permanent closure. In the final rule, these requirements must be consistent with the standards established by the EPA for a Yucca Mountain repository as required by the NWPA and the EnPA. The proposed NRC rule specifies the requirements for the safety analysis of the geologic repository to demonstrate compliance with the pre-closure performance objectives. The safety analysis would be a systematic examination of the potential hazards, which would ensure that all relevant hazards that could result in unacceptable consequences have been adequately evaluated and protective measures have been identified so that the repository will comply with the pre-closure performance objectives. The proposed rule also specifies the requirements for the performance assessment that must be used to demonstrate compliance with the post-closure performance objectives, including requirements regarding the characteristics of the reference biosphere and population group that must be used in the analysis.

The DOE proposed revisions to its site suitability guidelines in 1999 at 10 CFR Part 963 [7]. These revisions would align guidelines with the latest science and scientific analytical techniques for assessing repository performance. The proposed rule presents the criteria and methodologies that the DOE would use in determining whether a potential Yucca Mountain repository is likely to meet the applicable pre-closure and post-closure radiation protection standards. The proposed approach is consistent with the EPA's standards and the risk-informed, performance-based criteria proposed by the NRC. The DOE developed a draft notice of final rulemaking based on comments received and submitted it to the NRC in May 2000 for concurrence as required by the NWPA. NRC concurrence is required before the DOE can issue the guidelines as a final rule.

2. PERSPECTIVE ON A SAFETY CASE TO SUPPORT A POSSIBLE SITE RECOMMENDATION

A safety case comprises more than the results from a pre-closure or post-closure safety assessment. The idea of a safety case is to broaden the basis for confidence by decision-makers and the public in conclusions about safety. A safety case should cite multiple lines of evidence, or reasoning, beyond the results of a safety assessment to support the demonstration of safety, which includes compliance with applicable safety criteria. The multiple lines of evidence should show the basis for confidence in safety. To be most effective, such evidence requires information not directly used in the safety assessment. Although it is not always possible to cite information that is independent of the safety assessment, it may be possible to derive meaning from the information in ways that differ from its use or treatment in the safety assessment.

2.1. Audiences for a safety case for a possible site recommendation

Clearly, the case for site recommendation must be adequate for the Secretary to make a decision on whether or not to recommend approval of the site to the President, and for the President to consider the site qualified and submit the recommendation to Congress. There are, however, multiple additional audiences to be considered in developing the information and constructing a safety case (or safety cases) to support a possible site recommendation decision and a supporting evaluation of site suitability.

The technical basis for the evaluations and any decision must be credible to the general technical community. The views of regulatory and oversight bodies will also need to be considered. The NRC has stated that it will not comment on the DOE's site suitability evaluation, which requires evaluation of compliance with the applicable standards. The NWPA requires, however, that a decision by the Secretary to recommend the site consider the NRC's preliminary comments on the extent to which the DOE's information seems to be sufficient for inclusion in a license application. Multiple lines of evidence in support of the models and data used in the safety assessments, particularly the total system performance assessment (TSPA) conducted for the post-closure period, will be important in enhancing confidence in the basis for these assessments. Congress established the US Nuclear Waste Technical Review Board (NWTRB) in the amendments to the NWPA. The NWTRB reports its findings and recommendations to the Secretary and the Congress, and will express its views regarding the basis for a possible site recommendation. The NWTRB already has expressed concerns about several aspects of the technical basis for a site recommendation decision, including the need to develop multiple lines of evidence and argument for a safety case that are independent of the TSPA.

The public must also be considered in developing the documentation to explain the basis for any decision. The NWPA requires a hearing in the vicinity of the site to receive the public's views on the Secretary's consideration of a possible site recommendation. The States, in particular the State of Nevada and the local units of government, must be considered. Under the NWPA, the State of Nevada can file a notice of disapproval for any site recommendation submitted to the Congress. In that case, designation of the site for development of a repository would require Congress to pass a joint resolution of siting approval. The case supporting the site recommendation decision must be adequate to withstand challenge and support favorable action by the Congress.

2.2. Safety case in the context of the framework for site recommendation

The NWPA specifies the minimum information that must be considered by the Secretary in a comprehensive statement of the basis for a recommendation, including:

— A description of the repository with preliminary engineering specifications
— A description of the waste form and packaging, and its relationship to site conditions
— A discussion of data obtained in site characterization relating to the safety of the site
— A final EIS
— Preliminary comments of the NRC on whether the information developed by the DOE seems to be sufficient for inclusion in a license application
— The views and comments of any State or affected Indian tribe.

The evaluation of site suitability will contribute information based on site characterization data that relates to the safety of the site for development of a repository. The evaluation will be based on the methods and criteria in the DOE's site suitability guidelines. Consistent with these methods and criteria, the evaluation will include the essential elements of a safety case.

The comprehensive statement of the basis for a site recommendation will provide a safety case consistent with the information required by the NWPA. The comprehensive statement of the basis may also include other information the Secretary deems appropriate to provide perspective and enhance confidence in safety.

2.3. Case for the site suitability evaluation

The methods and criteria specified in the DOE's proposed site suitability guidelines require:

— Compliance with the applicable radiation protection standards established by the EPA
— Use of performance assessment for the post-closure evaluation, consistent with the NRC's proposed licensing criteria, including the following information:
  - Account for uncertainties and variability
  - Consider alternative models
  - Provide the technical basis for inclusion or exclusion of features, events, and processes, and for models used
  - Describe the capabilities of the individual barriers and provide the technical basis for this description
  - Conduct appropriate sensitivity studies
— Demonstration of a reasonable expectation that the potential repository will meet the applicable standards, considering both engineered and natural barriers.

The case for the suitability evaluation will consider a comprehensive technical basis, including multiple lines of evidence and argument:

— Field and laboratory data and analyses
— Information from natural and man-made analogs

— Results from expert elicitation and independent peer reviews (including an international peer review of the TSPA for site recommendation and an international peer review on waste package performance)
— Numerical modeling and analysis of pre-closure and post-closure system performance
— Evaluations of the capabilities of individual barriers to better understand the performance of the overall system
— Comparison of modeling results to applicable radiation protection standards for pre-closure and post-closure system performance.

3. ENHANCING THE CASE FOR A POSSIBLE SITE RECOMMENDATION

External reviews of the DOE program, most notably by the NWTRB, have identified concerns related to the technical basis for a possible site recommendation. These concerns are related to:

— Meaningful quantification of uncertainties and conservatism in modeling and performance analyses
— Progress in understanding the fundamental processes involved in predicting the rate of waste package corrosion
— Evaluation and comparison of a lower-temperature operating mode with the base case repository design (higher-temperature operating mode)
— Development of multiple lines of evidence and argument derived independent of the TSPA to support a safety case

The DOE is addressing the concerns that have been raised to enhance the case for a possible site recommendation. In particular, the DOE is considering additional lines of evidence, such as alternative analyses, information from natural analogs, use of simplified calculations, alternative performance indicators, and direct observations, as appropriate. As part of the basis for a site recommendation decision, the DOE will show that the scientific work underlying the decision is competent, technically defensible, and provides a sound basis for confidence in the safety case presented.

REFERENCES

[1] NUCLEAR WASTE POLICY ACT OF 1982, Public Law 97-425 (1983).
[2] NUCLEAR WASTE POLICY AMENDMENTS ACT OF 1987, Public Law 100-203 (1987).
[3] ENERGY POLICY ACT OF 1992, Public Law 102-486 (1992).
[4] 40 CFR (U.S. CODE OF FEDERAL REGULATIONS) PART 197, Environmental Radiation Protection Standards for Yucca Mountain, Nevada; Proposed Rule, 64 FR (Federal Register) 46975 (August 1999).
[5] 40 CFR PART 197, Public Health and Environmental Radiation Protection Standards for Yucca Mountain, Nevada; Final Rule, 66 FR 32074 (June 2001).
[6] 10 CFR PART 63, Disposal of High-Level Radioactive Waste in a Proposed Geologic Repository at Yucca Mountain, Nevada; Proposed Rule, 64 FR 8640 (February 1999).
[7] 10 CFR PART 963, Yucca Mountain Site Suitability Guidelines; Proposed Rulemaking, 64 FR 67054 (November 1999).

A FRAMEWORK FOR ELABORATING A GEOLOGICAL DISPOSAL SAFETY CASE: MAIN ISSUES TO BE ADDRESSED

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Abstract. International guidance on safety standards for the geological disposal of radioactive waste is being elaborated by IAEA. A comparison of experiences acquired in developing deep repository projects shows that many important issues related to the progressive building of confidence in the safety demonstration of such facilities are commonly addressed by the various organisations involved in radioactive waste management. However, there is still some discrepancies in defining the steps that form the staged elaboration of a safety case. This paper intends to propose a framework for defining the safety case in describing the main issues to be addressed and highlighting questions of consistency between former steps.

1. INTRODUCTION

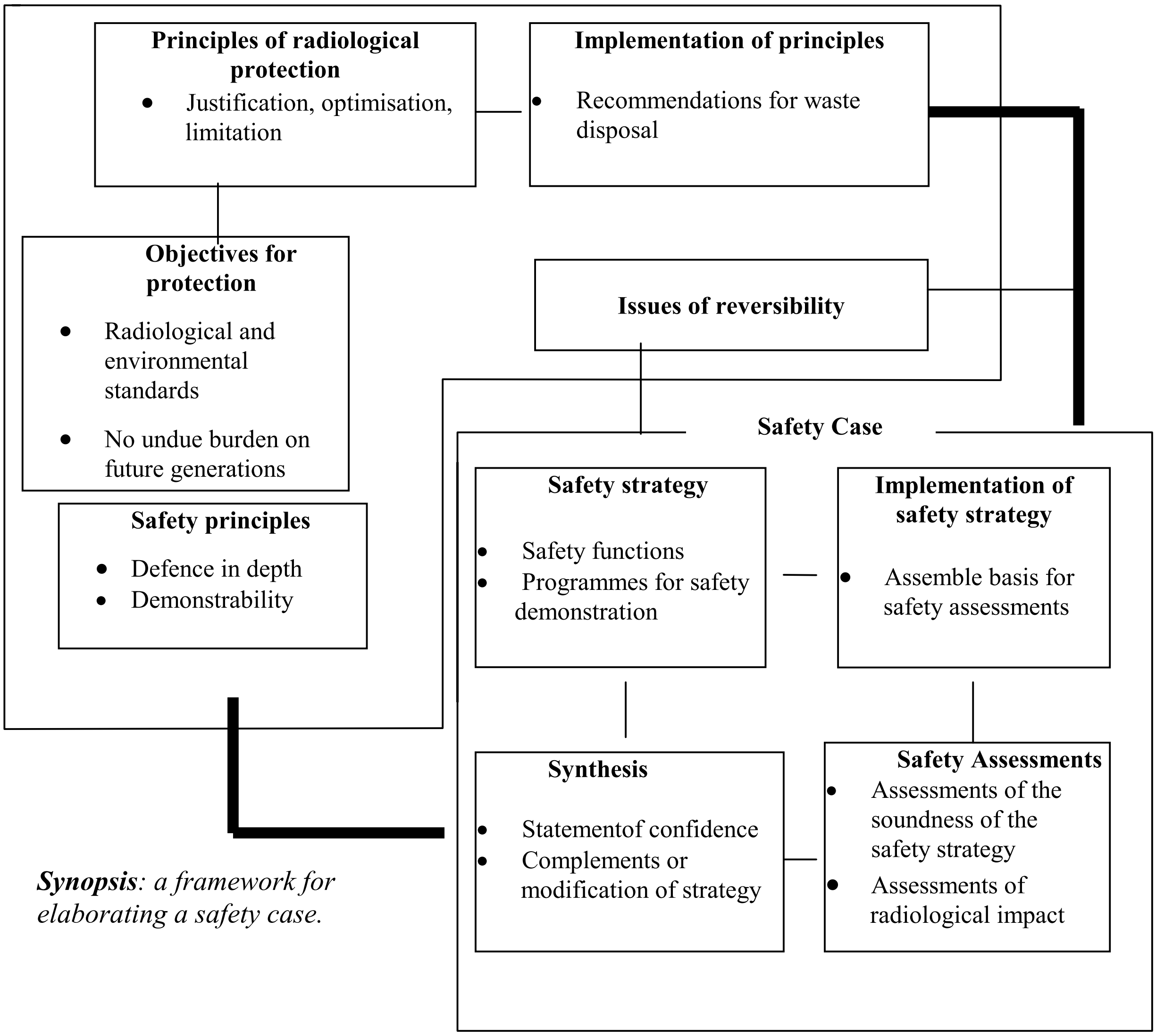
There is still little experience today for the elaboration, at the international level, of a precise set of regulatory requirements covering the safety of radioactive waste geological disposal, since almost all examples of national programmes carried out in this area are at the project level. However, efforts are made by international instances in elaborating a rationale for implementing the safety of a geological repository, rationale that may be applied at the various stages of development of a repository project.

The present paper intends to propose a framework for elaborating a safety case for the deep disposal of radioactive waste. This work uses information developed by international organisations, such as IAEA, AEN and ICRP as well as national guidance published in this area, i.e. the French Basic Safety rule n°III.2.f [1]. It also backs up on outcomes of working groups initiated by the French safety Authority (DSIN): a French–German group[1] that worked from 1997 to 1999 and a French–Belgium group[2] that started work in 2000.

The issues developed hereafter are resumed in the synopsis next page, which illustrates the various types and levels of information that form the proposed framework and how these relate to each other. As shown in the synopsis, the safety case is only a part of the overall framework, but is to be developed in consistency with the more general issues that are outside its precise scope. Therefore, various aspects of the framework are discussed here. One may also notice that, even if regulatory guidance and review addresses all aspects of the framework, the development of the safety case is the task of the operator, as the more general issues require primarily the involvement of regulatory bodies and possibly a consensus at international level. Following the scheme proposed in the synopsis, the various aspects of the framework concerning safety issues are shortly discussed and possibly illustrated by a few examples mostly taken from present experience of deep disposal project development in France. The paper does not aim at listing all issues to be addressed in a safety case, but rather at providing a support for collecting widespread information in this area, hoping that it helps

[1] French-German group: Direction de la Sûreté des Installations nucléaires (DSIN), Institut de Protection et de Sûreté Nucléaire (IPSN), Agence Nationale pour la gestion des Déchets Radioactifs (ANDRA), Bundesministerium für Umwelt, Naturschutz und Reaktosicherheit (BMU), Gesellschaft für Anlagen und Reaktorsicherheit (GRS), Bundesamt für Strahlenschutz (BfS).
[2] French-Belgium group: DSIN, IPSN, ANDRA, Agence fédérale de Contrôle Nucléaire (AFCN), Association Vincotte Nucléaire (AVN), Organisme National des Déchets Radioactifs et des matières Fissiles enrichies (ONDRAF).

in selecting issues that may already be resolved internationally from those requiring further experience to be acquired.



***Synopsis**: a framework for elaborating a safety case.*

2. OBJECTIVES AND PRINCIPLES GUIDING THE ELABORATION OF THE SAFETY CASE

2.1. Radiological and environmental protection objectives and principles

International guidance is given on this area in particular in IAEA Basic Safety Standards [2] and ICRP publication 81 [3]. Whether radiological and environmental protection issues need more discussion for enhancing international agreement is not in the scope of the present paper and therefore not addressed any further here. One may just point out that ICRP stresses that the following of sound technical and managerial principles and the quality of data and analysis supporting the safety assessments of a repository are important factors for judging whether radiological protection requirements can be considered satisfied. The issues discussed hereafter concern precisely the definition and implementation of such principles in the safety case.

2.2. Safety principles

Safety principles aim at giving general guidance to the development of safety strategies and assessments for deep disposal of radioactive waste. They concern all phases of development of a repository project, from the conception (site selection and design) to the realisation and operation of the repository. A careful application of these principles may allow protection objectives and principles to be satisfied, through:

— the implementation of a cautious approach for design, providing solutions against plausible causes and consequences of disposal dysfunction,
— the creation of safety margins,
— the build up of confidence in safety demonstration,

Despite questions of terminology, which may need harmonisation, the issues grouped hereafter in notions of "defence in depth" and "demonstrability" are frequently addressed by organisations involved in repository development. Thus, an international consensus seems foreseeable on the basic safety principles that should apply to deep disposal of radioactive waste.

2.2.1. Defence in depth

Applied for designing an operating nuclear power plants and fuel cycle facilities, defence in depth is defined in IAEA Safety Glossary [4] as *"a hierarchical deployment of different levels of equipment and procedures in order to maintain the effectiveness of physical barriers placed between a radiation source or radioactive materials and workers, members of the public or the environment, in operational states and, for some barriers, in accident conditions. The objectives of defence in depth are:*

— *to compensate for potential human and component failures,*
— *to maintain the effectiveness of the barriers by averting damage to the facility and to the barriers themselves,*
— *to protect the public and the environment from harm in the event that these barriers are not fully effective."*

This definition fully applies to waste disposal during phases of operation of the facility (construction, emplacement of waste, possible retrieval of waste, operations of closure). Deep disposal's specificity is however that actions aiming at maintaining the effectiveness of physical barriers are excluded after closure of the facility. Over the long term, defence in depth can therefore only rely on "passive means", i.e. on design features, in order to achieve the safety of waste disposal. In that sense, the defence in depth objectives, as formulated above, cannot be assigned *in extenso* to deep repositories. Nevertheless, the fundamentals of defence in depth can still be derived for deep disposal. In this respect, a widely agreed approach is to design a repository upon a set of requirements that constitutes the bases of a **"multi-barrier concept"**. Accordingly, it is recommended that a repository be conceived by means of several barriers so that the failure of one barrier does not impair seriously the safety of the repository. This means implicitly that:

— the design of the repository must account for an analysis of the possible causes and consequences of failure of the barriers,

— in consistency with defence in depth, design should be such as to compensate for barrier failure, by averting as far as possible the causes of failures and mitigating their consequences,
— several components of the repository must intrinsically oppose to the dissemination of radionuclides, so that protection objectives can be matched with margins (accordingly, BSR III.2.f recommends to adopt a "prudent approach, consisting at selecting or designing the barriers to be effective as is reasonably possible, taking into consideration its role in the overall safety of the repository as well as present knowledge, the available technology and economic factors").

These elements constitute a safety principle that should be accounted for in elaborating a safety strategy (cf. § 3.1).

2.2.2. Demonstrability

Demonstrating the safety of deep geological disposal faces issues to be resolved that are not encountered for other facilities, since disposal must ensure protection against the dissemination of radionuclides over periods that do not allow to back up demonstration on control nor available experience in industry. This peculiarity must be accounted for from the very beginning of project development so as to come up with design and safety strategies that facilitate demonstration over the long term. Accordingly, the following requirements should apply when designing a repository, i.e.:

— the design should rely on **robust** feature, with respect to the long time period over which the functions played by the various repository components must be ensured. It is therefore believed that the “durability” required for a given function of a component is better obtained if the component is robust, in the sense that its performances are not sensitive to the variations in time of its environmental conditions. Robustness is thus helpful in providing confidence that safety can be demonstrated, and the selection of a repository component should be a balanced decision justified on the expected performances of a component on one hand, and its ability to keep performing in the various conditions of environment expected over long time frames on the other hand;
— accounting for the durability of performances that may be required from repository components, a particular attention should be paid to the **technical feasibility** of the components, i.e. to the qualification of techniques and controls ensuring an appropriate quality of realisation such that a good confidence can be obtained in component ability to fulfil their functions;
— only poor confidence will be credited to safety demonstration if too complex processes and interactions influence the performance evolution of the components. As far as possible, a **simplicity** of design should be looked for, so that the components evolution can be evaluated within a field of **consolidated knowledge** of the data and processes that originate this evolution.

The issues above are addressed in IAEA draft version of safety requirements developed for geological disposal of radioactive waste [5]. In particular, authors recommend that "*a geological repository be sited, designed, constructed and closed so that the natural and engineered features and characteristics on which the long term safety depends are reliable over suitably long time periods. That is the key features should be intrinsic properties or sufficiently reliable, and processes on which safety depends should be well understood over the required time scales*". They also point out that *"the construction of a repository is a complex technical undertaking and will be constrained by rock conditions and the available* *underground excavation and construction techniques. The construction should be arranged in timing and quality so as to ensure the expected safety functions of the geology and engineered barriers as realized"*. International agreement on these demonstrability issues seems therefore to progress.

2.3. Issues of retrievability

Issues of reversibility have actually been raised and discussed during the past few years in many countries involved in HLW management. Reversibility of disposal is not required for safety purposes but relates to different ethical views recently expressed on the duty of present generations towards future ones, resulting in two opposite opinions with regard to the question of waste management. One side considers that the present generations should provide and implement a definitive solution of waste management so that the future generations do not bear the “undue burden” of going on with the management. The other side considers that the right of future generations is to be free of their own choices of waste management and that the duty of present generations is to take no decision that may impair this freedom. According to this last opinion, a disposal should be reversible so that the waste emplaced can be retrieved to be possibly managed in another way.

The definition of a "reversible" repository needs however to be clarified and requires to be possibly agreed upon at international level. IPSN position on this issue is related to the possible role of reversibility in the process of implementation of a repository. Accounting for the debate on ethical issues of waste management, a reversible disposal could be considered as a solution consisting in allowing future generations to take benefit of the experience of present generations in designing a HLW repository, without impairing their freedom of choice. The repository should therefore be designed so as to ensure its long term safety (meaning that all aspects of safety have been studied and that appropriate techniques for closing the repository have been qualified), but allowing future generations to be free of making basically three choices: retrieve easily the waste if another way of managing them is preferred, extent the period of reversibility through possible maintenance operations, close partially or completely the disposal. Thus, a reversible phase of disposal should allow time for taking appropriate decision, and should provide in particular elements of verification that the behaviour of the facility is such as expected which progressively enhances the confidence that one may have in the safety assessment of the repository. However, questions on the consequences on safety of a prolonged phase during which the repository is left opened should be treated with particular care, and should contribute to determine the duration of the reversible period. Among them are in particular the mechanical stability of excavated zones as well as the evolution of package integrity in a possibly oxidising environment. More generally, the provisions made for reversibility purposes (architectural features, equipment for maintaining voids in the vaults for allowing waste handling, equipment for extended ventilation) should be weighed against their consequences on pre and post closure safety.

It is now required in most countries to study reversible disposal. Therefore, it should be included in the safety case and addressed in particular with regard to its incidence on design and operation of the disposal and related short and long term issues.

3. ISSUES TO BE ADDRESSED IN THE SAFETY CASE

NEA publication on “confidence in the long-term safety of deep geological repositories” [6] has in particular highlighted that confidence building in the safety of a repository is an incremental process which should be made explicit within a safety case. A safety case is to be provided at various steps of repository development, but should be structured, whatever stage of development is considered, on the same basic elements which include the definition of a strategy, the assembly of relevant information models and methods, performance assessments and statements of confidence in the safety of the repository. These notions are broadly addressed at international and national level but it appears that they are given somewhat different meaning among the organisations involved in repository project development and that an harmonisation of terminology should be looked for in this area. The elements of information given hereafter are therefore proposals to be discussed and only reflect the author's position on the subject.

Repository development should rely on an **iterative and structured approach** allowing design, knowledge acquisition and safety assessment steps to develop in a fully complementary way, so as to progressively propose a conception consistent with defence in depth and demonstrability issues and a clear management of uncertainties. It should allow to build confidence in safety but also to go back on design at every important step of project development. The safety case should report on the progresses made at each step of project development towards the safety demonstration of the repository, and should comprise:

— a description of the **safety strategy**, which guides any particular step of development of the project through the definition of the safety functions assigned to the various components of the repository and through the definition of the programme of investigation needed to confirm that these functions are effective,
— a description of the data, experiments and models acquired and developed during any particular step of project development, which constitute the main tasks of **implementation of the safety strategy**,
— description of the **safety assessments** performed, aiming at verifying the soundness of the strategy proposed and at verifying that protection objectives are fulfilled,
— a **synthesis** of the results obtained giving the main orientations to be followed during the next step of project development.

These four issues are discussed hereafter.

3.1. Safety strategy

The safety strategy for conceiving and realising a safe repository is of the responsibility of the operator. It should precisely describe the **safety functions** and associated favourable characteristics upon which the overall safety of the repository relies, the system components that ensure these functions and the situations over which these functions and characteristics are defined. These constitute the major elements against which the conception of the repository is justified and assessed and must be developed in consistency with the safety principles mentioned in § 2.2. The safety strategy must also comprise a clear definition of the programme of investigations to be developed in order to confirm that the selected system components have the capacity to ensure their function over the required period. Therefore, the features and processes relevant to such evaluation and which form the **elements of demonstration** over which the programme of investigation backs up must be described in the safety strategy.

It should be acknowledged that several strategies may be proposed depending on the availability of sites, the type and volumes of waste to be disposed of, and socio-economic factors, so that consensus on a single strategy to apply have a very poor chance to be obtained.

A comparison of the strategies developed in various countries can however be useful if grouped into categories of similar factors.

3.1.1. Definition of safety functions

The **safety functions** that are to be attributed by the operator to the components of the system are the key elements of the strategy, since they build the rationale upon which a sound design can be implemented. Several functions can be attributed to a same component, depending in particular of the phases and the scenario considered, the time frame over which a function must be effective, and the dependency to other components of the system. The functions of which relative importance for the safety of the repository are to be discussed can be grouped in three categories:

— the functions directly contributing at **reducing the fluxes of radionuclides** and other substances from the repository to the biosphere; among these are:
— functions of confinement, aiming at a total containment of activity in a given component within a given period of time. These may be obtained for example through the tightness properties of waste envelopes,
— functions of retardation of activity migration, allowing a reduction of fluxes by decay, and that may be obtained for example through slow groundwater flow properties, retention properties or matrix diffusion properties,
— functions limiting the concentration of radionuclides and substances in transport vectors, that may be obtained for example through low release rates properties of waste packages, through a limitation of the solubility of radionuclides and chemical substances in barriers, through the averaging of peak releases by diffusion and dispersion or through the dilution of releases,
— the functions contributing at **protecting the system** against external factors, that may be obtained through for example a limitation of access possibilities by a minimum depth of the repository or through a favourable location of site away from valuable underground resources or in desert areas,
— the functions contributing to the **preservation of favourable conditions** and to the robustness of the system; these are in particular functions of components aiming at providing the best conditions for other components to play their role in flux reduction, among this group of functions are for example, the restoration of system disturbances such as the restoration of mechanical stability, the preservation of favourable hydraulic and geochemical features, the restoration of adequate levels of heat dissipation.

A selection of **components** should then be made upon the choices made by the operator of the safety functions on which the system should be conceived, accounting for the various phases of the life of the repository. It is a key aspect of the strategy to clearly select or attribute safety functions to the system components and provide a hierarchy of these functions so as to identify the main favourable characteristics over which the components should be assessed or designed. One may note that IAEA safety glossary define barriers as "*a physical obstruction that prevents or inhibits the movement of people, radionuclides or some other phenomenon (e.g. fire), or provides shielding against radiation*" Following this broad definition, the safety glossary defines intrusion barriers as "*components of a repository designed to prevent inadvertent access to the waste by humans, animals or plants*" and multiple barriers as "*two or more natural or engineered barriers used to isolate radioactive waste in, and prevent migration of radionuclides from, a repository*". According to this definition, there is rather a wide agreement to consider the **geological formation** (consisting of the host rock formation and possibly additional geological units overlying or underlying the host rock), and the **waste**

packages (including waste form, matrixes and envelopes and possibly canisters) as being barriers, since they usually ensure functions of protection against intrusion and functions of reduction of radioactive dissemination. There is however large differences between countries involved in repository development concerning the definition of the performances required from these barriers and the time frames over which they must ensure their functions. Another source of differences between national approaches concern the role of **engineered components** of the repository (mainly buffers, seals and backfill). Whether these should ensure functions that are **redundant** of geological formation and waste packages or primarily play a **complementary** role (i.e. provide favourable conditions for ensuring that barriers are effective during the required period) is not settled and strongly depends on the type of geological formation and waste packages considered in repository development. Finally, the **situations** against which the functions and components of the system should be defined are also important elements of the strategy. Throughout the development of the project, the operator will have to make choices in selecting these situations (barriers and components dysfunction, external events, etc.) and provide a hierarchy of these. These choices will have to be clearly reported throughout the project. It is also important to address situations, not necessarily linked to the long term safety of the repository, but against which the design is likely to be strongly influenced. This concern in particular the provisions made for a safe operation of the repository and for the possible retrieval of waste.

In addition to these elements of strategy, a particular attention is to be given to the definition of **architectural features**, such as galleries, vaults and shafts, of which geometries, relative arrangements and emplacement in site can be of importance for safety. A safety strategy should therefore describe how the chosen elements of architecture are in favour of the overall safety of a repository together with the engineering methods needed to realise these elements (incidence of the techniques of construction used, incidence of the support equipment used).

The French Basic Safety Rule III.2.f provides an illustration of regulatory guidance given on former elements of a safety strategy. The **geological barrier** must ensure both functions of protection of the waste against external events and containment of radionuclides. It plays a key role over the long term since it is assumed that "*after the decay of the greater part of radioactivity contained in the waste, the geological barrier (and the shaft sealing materials) must be able to ensure containment alone*". The effectiveness of the geological barrier functions backs up on requirements of stability, prevalence of low permeability and hydraulic gradients, favourable geochemical, mechanical and thermal properties, a minimum depth requirement, and absence of underground resources, in consistency with defence in depth and demonstrability principles mentioned in § 2.2. The **waste packages** must ensure protection against dissemination of radionuclides during the phases prior to the closure of the repository. They should also prevent this dissemination after closure of the repository in case of a short circuiting of the geological barrier and during a period allowing decay of short and intermediate lived radionuclides. There are also requirements for **complementary roles** played by the waste packages in the sense that they should ensure sufficient durability of the properties of the materials in which they are emplaced (in particular through the respect of temperature and irradiation limits, minimisation of volumes of voids and absence of products liable to increase the mobility of the radionuclides in containment barriers). In the French approach, the **engineered components** are mostly viewed as means for restoring initial properties of the geological barriers and for minimising perturbations (prevent excavations to constitute preferential pathways, prevent collapsing, reduction of the mechanical stresses, removal of heat, etc.). They should not have a significant detrimental effect on the containment performances of the geological barrier or the waste packages. Finally, broad guidance is given on **architectural features** in requiring that the modules of disposal be installed within a medium free of major cases of heterogeneity (sedimentary rock) and at sufficient distance from the surrounding aquifers and that the shafts be designed and positioned so as to limit water circulation and with the objective of effective sealing. It is also required that disturbances resulting from excavation of the facilities be minimised.

3.1.2. Elements for demonstrating safety

The second important aspect to be developed in a safety strategy is to provide a justified programme of investigation and studies to be carried out for progressing in the safety demonstration of the repository, in consistency with the proposed sets of safety functions and components chosen in a particular stage of project development. The programme must define the main issues of demonstration to be dealt with during a stage of project development, and the means to be implemented for safety assessment purposes. It must also be put in perspective within the overall project so that the expected progresses in building confidence in safety demonstration during a particular step be reasonably defined in time. Thus, a hierarchy of the issues to be treated must be provided and justified in the safety strategy. Accordingly, the main aspects to be reported would be:

— a clear definition of the main characteristics of the repository components and the processes on which the safety functions rely,
— a description of the "state of the art" of knowledge on these characteristics and processes and their possible evolution in repository conditions,
— a selection of the issues to be further investigated so as to enhance confidence in safety demonstration up to a level considered satisfactory,
— a description of the researches, tests and modelling to be implemented on these issues within a particular stage of repository development, with due consideration in providing results that are representative of repository conditions and of situations identified by previous safety assessments as deserving careful treatment,
— an argumentation allowing to put a particular stage of repository development in perspective with the overall project so as to clearly illustrate the strategy used for progressively resolve uncertainty and undetermination issues, and match possible regulatory requirements related to the phasing of the project.

3.2. Implementation of the safety strategy

The implementation of the safety strategy comprises reporting on the programme of investigation carried out during a particular step of project development, in consistency with the issues and objectives defined in the safety strategy. It should provide all the necessary elements for safety assessments of the repository, and therefore give a detailed description of:

— the design of the repository, developed in consistency with the safety strategy, including site data, waste packages volumes and characteristics as well as amounts and characteristics of the materials introduced in the repository (engineered components),
— the data obtained on the characteristics and processes selected in the safety strategy, and that should be provided in quality and number compatible with the needs for obtaining information representative of repository conditions,
— the results of the experimental tests together with the processes modelling conducted in order to provide the necessary information and tools for quantitative evaluations,

— the results of the technological experiments carried out to demonstrate that a quality of engineered components fabrication and emplacement compatible with the required efficiencies of functions can be obtained.

The operator should make sure that all records are kept of the detailed description of the means and hypothesis used to obtain data and models, which could be needed at any time for safety assessment purposes.

The implementation of the safety strategy is a most important phase of project development, not only because it provides the necessary input for safety demonstration but also because it needs important time and resources allocation. The hierarchy of the issues to be treated and the adequation of the means of investigation implemented to treat them are therefore key issues of the safety case. Thus regulatory guidance should be looked for on this area in addressing questions to be resolved for a sound implementation of the strategy. Among the main difficulties commonly raised are the up scaling of results in time and space and the implementation of representative tests and models. How site investigation should be phased and organised so as to obtain information that are valid at repository scale? What methods are available to resolve the bias introduced by the acceleration of processes in experimental tests or the extrapolation of results in time (iteration between data acquisition and modelling, use of "envelope" but representative hypothesis in experimental design and models, etc.)? These are areas where the provision of guidelines obtained through the analysis of international experience would certainly be needed.

3.3. Safety assessment

The safety assessment of a repository aims at evaluating the soundness of the safety strategy on one hand and at verifying that the repository performs such as adequate levels of protection of man and environment are reached on the other hand. It is during this step that the “global” performance of the repository is evaluated against plausible situations (scenarios). This requires a policy for scenario definition and development and for integrating all pertinent information into a sound modelling. In return, safety assessments provide an input for the treatment of uncertainties and contribute to provide a hierarchy of the studies deserving particular attention and that should be implemented in the next stage of project development. It should also provide input for modifying or complementing the strategy, and confirm, in advanced stages of development, that the application of specified parameters lead to adequate levels of protection. The realisation of safety assessments requires:

— to assemble all qualitative and quantitative information obtained in particular during the phase of implementation of the safety strategy in order to determine a plausible set of scenarios of repository evolution,
— to test the reliability of the safety functions against these scenarios through performance assessments using in particular the "process/subsystem" modelling that has been possibly validated during the phase of implementation of the strategy,
— to use integrated performance assessment (and sensitivity analysis), enabling to evaluate the fluxes and concentrations of radionuclides released in the various parts of the repository, for various time frames and for the set of scenarios determined previously, so as to evaluate quantitatively the pros and cons of the proposed safety strategy, and highlight the various roles played by the barriers in time with regard to radionuclides releases and migration,
— to use also integrated performance assessment to verify that the protection standards can be satisfied.

An issue that would probably deserve reaching consensus at international level concerns the use of **realistic** and **conservative** hypothesis for scenario development. Taking example from the French approach to illustrate this issue, scenario definition and development can result from a process of selection of plausible situations, issued from an analysis of repository conditions and evolution, and a grouping of those situations in families. For each of these families, a possible “envelope” scenario is defined and assessed in a deterministic way with appropriate tools. It is however needed to **avoid over simplification** and perform reasonably realistic calculations in order to derive pertinent information for evaluating the soundness of the safety strategy on one hand, and on the other hand to allow the definition of a pertinent “envelope” scenario through sensitivity analysis and appreciation of uncertainty around a "best estimate" hypothesis. Therefore, a precise input from system analysis and testing is required for developing a scenario, the use of “stylised” description being appropriate for covering areas of which uncertainty cannot be reduced (biospheres, scenarios beyond the period of geological stability). Nevertheless, once sufficient safety assessment exercises have been realised so as to confirm that the safety strategy is pertinent, the specified characteristics and properties of the various components of the repository will clearly have to be determined upon **conservative** hypothesis, in consistency with the envelope scenarios defined, so as to ensure that protection standards are satisfied, with impacts kept "as low as reasonably achievable". International consensus on this issue should be looked for or possibly reaffirmed. It would also be helpful to elaborate common guidance on developing "**conventional**" scenarios where stylised descriptions can only be provided or for situations that are not or poorly related to site and design specificities. This concerns in particular biosphere hypothesis and intrusion scenarios.

Concerning integrated performance assessment, modelling efforts must be made to integrate all pertinent information per scenario into a model allowing the calculation of the releases of radionuclides from the repository. The complexity of the processes and coupling of processes involved in the assessment of a possible evolution of a repository requires simplification for integration in performance assessment tools (IPA tool). One way to proceed is to carry out sensitivity analyses on the assessment with an IPA tool so as to select the subsystem parameters and processes that require to be more specifically accounted for in the modelling. On these subsystems, process modelling may be performed together with a subsystem sensitivity and uncertainty analysis so that a range of possible variation of “global” parameters, used in IPA tools, can be determined (for example mechanical calculations determining the sizes and permeability of the damage zone, geochemical calculations to derive Kd and solubility values in various conditions). If simplification is required, it must however be fully justified, i.e. the representation of the properties of a subsystem can be rigorously averaged. If not, the subsystems and necessary coupling of processes should be accounted for explicitly in the modelling. Guidance on the methods for assembling relevant data and models into IPA tools so as to obtain credible quantitative assessments is also an issue that would deserve discussion at international level. This raises in particular the question of the "**validation**" of integrated assessments which is to be provided *pro parte*, on subsystems or analogues, since a post-verification of the results of integrated assessments cannot be done attended the time frames considered.

The results of impact assessment will have to be expressed in such a way that enough elements are provided for judging of impact acceptability by the various instances. The first requirement is to provide a treacability of the results obtained by making all assumptions used for calculations clear and justified. The second requirement is to provide the necessary elements describing the impact (individual effective dose, size of critical group, duration of exposure, impact of toxic substances eventually released). Finally, it may be of interest to use other indicators for judging of the impact, such indicators belonging to two broad categories: those which allow to enhance confidence in the calculations (for example, source terms, fluxes through barriers, dilution factors or fluxes in the biosphere can be compared to possible analogues), those that may complement dose calculations (for example, a comparison with U-ore impacts may be useful when U and daughters become the main potential contributor to the impact, and in particular, when too large uncertainties on the possible state of the repository do not allow a pertinent dose assessment). The IPA tools should account for the calculation of such complementary results. Integrated performance assessments are also important elements for enhancing confidence in the pertinence of the proposed design and, since it integrates information, is a key step for communicating the results. The expression of results from the performance assessment is therefore an aspect that deserves attention.

3.4. Synthesis

The last aspect of the safety case is to provide a synthesis of the work performed during a particular step of repository development, aiming in particular at giving a **statement of confidence** in the safety demonstration of the repository. Such statement should describe how the work performed has allowed to enhance confidence in the demonstration that safety functions are effective, that disturbances of the repository system are adequately prevented or minimised, that the proposed multi-barrier concept shows sufficient robustness with regard to the durability required for repository components, and that technical feasibility is ensured. This statement must therefore resume a new state of the art on the elements of a safety demonstration. It should also highlight the issues that need further treatment in order to enhance the credibility of assessments and to narrow the range of possible variations of input and results until sufficient confidence in the assessment is obtained. The synthesis will thus define the main guidelines for complementing or modifying the safety strategy, and address whether remaining uncertainties should be reduced by way of research work (to compensate lack of data or understanding of the processes) or by use of technical means (for example, a change of design allowing to avoid considering a process that would have consequences of too much complexity to enable credible assessment). When uncertainties arise that need obviously important investigations to be resolved and that concern important elements of a safety strategy, it seems prudent to recommend keeping **alternative design options opened**, so that technical solutions are available if research work fails to demonstrate the soundness of the followed strategy.

It should also be added that confidence building may also rely on the monitoring of repository behaviour during operational and reversibility phases allowing the control of parameters that indicate that the components behave as expected. According to the functions and characteristics assigned to the components, a list of "controllable parameters" reflecting the effectiveness of the latter should be provided by the operator, together with the technical means of control.

Finally, care will have to be given to provide information that is, when possible, consistent with public preoccupations. Some questions often raised are naturally within the scope of the safety case such as concerns expressed on the credibility and completeness of the scenarios, the treatment of intrusion scenarios, the position of discharge points. Others are not necessarily in the scope of the safety case but however concern issues that would probably deserve to be accounted for in it. These concern for example the impact of the repository on local environment, the total capacity of a particular site to dispose of the radioactive waste production in case of continuation or further development of nuclear industry, the inventory of existing radioactive waste and the rationale and consistency of the various types of management they receive.

The collection of international experience in addressing such issues can be helpful to possibly complement the recommended contents of a safety case.

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A.4. SAFETY INDICATORS

NATURAL SAFETY INDICATORS AND THEIR APPLICATION TO REPOSITORY SAFETY CASES

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Abstract. Radiological dose and risk are the standard end-points calculated in all performance assessments. Their calculation requires, however, assumptions to be made for future human behaviour. To complement dose and risk, other safety indicators have been suggested which do not require such assumptions to be made. One proposed set of safety indicators are the concentrations and fluxes of naturally-occurring chemical species in the environment which may be compared with the performance assessment predictions of repository releases. Such comparisons can be valid because both the natural and repository species would occur in the same system and their transport behaviour would be controlled by exactly the same processes at the same rates. Although simple in concept, there is currently no consensus on the most appropriate comparisons to make or on the interpretation of such comparisons. A number of national and international research projects are evaluating this proposed approach, including an IAEA Co-ordinated Research Programme. These projects suggest that that the approach appears to be workable and that it may be a valuable component of a safety case, complementing the dose and risk presentations. Further work is, however, necessary to develop the approach to a level where it may be confidently applied in further performance assessments in a consistent and methodical manner.

1. BACKGROUND

The generally accepted best practicable option for dealing with long-lived radioactive waste is final disposal in engineered repositories located deep underground in suitable geological environments. The safety of any proposed radioactive waste repository will be evaluated in a performance assessment (PA) which models in a simplified but adequate fashion, the many processes which may lead to the release and transport of radionuclides away from the repository, through the host rock to the surface environment and eventual uptake by humans. Post-closure repository performance is generally quantified in terms of radiological dose or risk to humans, with safety being determined by whether the calculated exposure values are consistent with predetermined target criteria which are deemed to represent acceptable radiological hazards.

Radiological exposure based safety indicators have been widely adopted by licensing authorities, and dose and risk have been universally calculated in all recent PAs. Despite their wide spread usage, however, there are various disadvantages with using dose and risk, such as:

(1) The calculation of dose/risk requires gross assumptions to be made for future human behaviour and the evolution of the surface environment (i.e. the human exposure pathway).

(2) The calculation of dose/risk to humans provides limited information on the wider potential environmental impacts arising from repository releases.

The mathematical derivation and presentation of dose/risk calculations, and the underlying fundamental radiological concepts, are difficult to understand by non-technical audiences.

Given these limitations, the use of safety indicators, other than dose and risk, to assess and confirm the safety of radioactive waste repositories has been suggested repeatedly through the years [e.g. IAEA, 1994]. This report concluded that a safety case can most effectively be made by the combined use of several safety indicators, while recognising that dose and risk remain the most fundamental indicators of safety. One proposed set of safety indicators, which is discussed in this paper, are the concentrations and fluxes of naturally-occurring chemical species in the environment (known as *natural safety indicators*) which may be compared with the PA predictions of repository releases.

At the outset, it should be stated that it is not the intention that these natural safety indicators should be considered as replacements for dose and risk. Rather, this paper discusses their potential use to complement dose and risk in a comprehensive safety case, to provide multiple lines of evidence for repository behaviour and safety.

2. CONCENTRATIONS AND FLUXES

All rocks, groundwaters, surface waters soils etc. naturally contain a wide range of radionuclides, such as U-238, Th-232. K-40 and Rb-87. These naturally-occurring radionuclides will be present in the environment around a repository at the time it is closed. At future times, if any radionuclides migrate from the engineered barriers into the host rocks and groundwaters, these repository-derived radionuclides will exist in the same system as the natural radionuclides, and will be subjected to the same processes causing their mobility (or retardation), acting in the same location and at the same rates (Figure 1). As such, it is conceptually simple to compare the PA predictions for the concentrations and fluxes of repository-derived radionuclides with the measured (actual) concentrations and fluxes of naturally-occurring radionuclides in the same environment, to provide a natural context for the interpretation of the associated hazards.

Put simply, this means that it should be possible (provided relevant data are available) to compare the predicted total repository-derived radioactive flux (e.g. in Bq/yr) crossing the geosphere–biosphere interface in the vicinity of the repository with the natural equivalent. The fundamental logic behind this comparison is that, if the flux to the biosphere from the repository is not significant compared with the natural flux from the geosphere, then its radiological significance should not be of great or priority concern. Numerous comparisons between repository and natural species concentrations and fluxes are theoretically possible, and decisions will need to be made on a case-by-case basis to determine the most appropriate comparison(s) to meet the objectives of the safety assessment in hand.

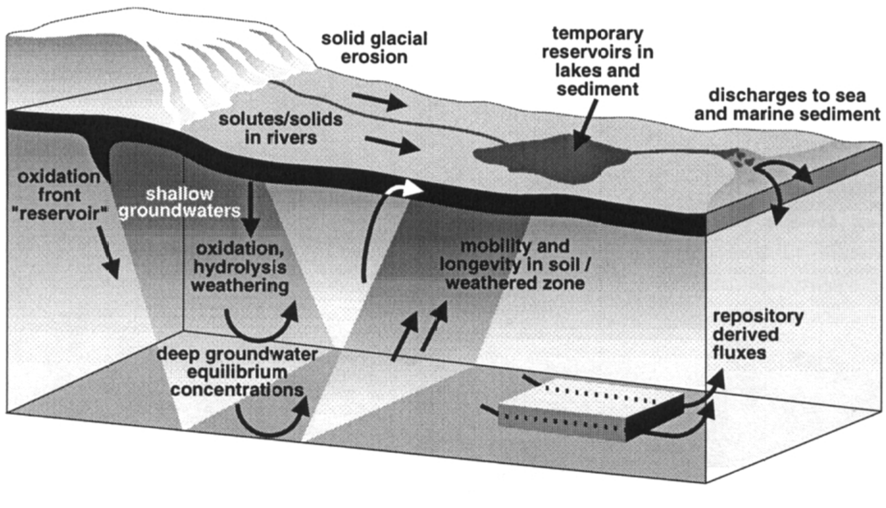

FIG. 1. Illustration of the comparison between the concentration and fluxes of repository-derived radionuclides and their natural counterparts, occurring in the same environment around a repository.

3. APPLICATION OF NATURAL SAFETY INDICATORS

A prerequisite of the natural safety indicators method is knowledge of the distribution and abundances of naturally-occurring species in the rocks, groundwaters, sediments etc around a repository. Average concentrations of many naturally-occurring elements and nuclides in surface and near surface media are relatively well known for a wide range of different environmental conditions and with a good geographical coverage, at least for certain countries [e.g. Miller et al., 2000]. Within the field of geochemical knowledge, the database on naturally-occurring radioelements is particularly well developed. Many of these data, however, relate to wide spatial averages and, thus, when looking at a specific candidate repository site, pre-existing geochemical data may not be adequate or sufficiently abundant either for the site itself or the surrounding area. In this case, the site-specific geochemical data will need to be obtained from the repository site characterization programme.

When considering fluxes, these cannot be measured directly and must be derived from concentrations on the basis of other parameters, usually the relevant process rate for the mechanism causing mobility (e.g. an erosion rate or groundwater flow rate). As such, it will be necessary for a site characterization programme also to measure relevant process rates.

If the concentrations and fluxes of the naturally-occurring radionuclides are known, they can be compared with the calculated repository releases in either a quantitative or qualitative manner. At the very basic level, simple qualitative comparisons can be made between concentrations of repository-derived radionuclides and their natural counterpart at the repository site (e.g. in the soils and groundwater). No conclusive judgment on the acceptability (safety) of the repository releases to the soil may be made, leaving the comparison to 'speak for itself'. A particular issue for the comparison is the spatial scale of the system under consideration. The most appropriate spatial scale may often be the catchment or watershed in which a repository is to be located (rather than the local site itself) because this generally defines the limits of the groundwater flow system, and the erosional and depositional processes. The catchment scale, therefore, essentially defines the extent of the system in which both the natural and repository species are moving together and may be the most direct comparison. Since one of the objectives is, however, to give 'context' to the repository releases, it is also useful to compare concentrations to the widest natural range which might occur regionally or even globally. In this way, it can be demonstrated how the repository releases compare to the full spectrum of possible natural abundances, and this may be readily presented in graphical format.

The more quantitative approach is to expand the simple comparisons described above to include some measure of safety (an appropriate hazard index), against which the comparison is judged to determine if the repository releases are acceptable or not. The latter approach is potentially very powerful if relevant hazard indices can be identified. Various options have been suggested including drinking water standards (e.g. from the WHO) and Annual Limits on Intake (ALI's, e.g. from the ICRP).

Although simple in basic concept, the natural safety indicators methodology is not, however, without its own drawbacks. Foremost amongst them is the fact that not all of the nuclides found in radioactive waste occur naturally and, thus, direct one-to-one comparisons of concentrations and fluxes are not always possible for all chemical species of interest to PA. A number of approaches may be adopted, however, to resolve this problem. One idea, which is applicable only to alpha emitters, is to assume that non-natural alpha emitting radionuclides have the same radiotoxicity as naturally-occurring ones. Other drawbacks to the methodology include:

— the lack of appropriate data on measured concentrations and fluxes of naturally-occurring radionuclides;
— the lack of internationally agreed forms of comparisons between repository and natural radionuclide abundances; and
— the absence of internationally agreed hazard indices against which to evaluate the comparisons.

4. INTERNATIONAL PROGRESS

Despite these limitations, a number of projects have been undertaken in the last few years to test the concept of using natural safety indicators in a safety case and to develop the approach further. Foremost amongst these projects in the IAEA Co-ordinated Research Programme (CRP). This has the overall objective of developing international consensus on the use of natural safety indicators. The specific research objectives are focused on resolving some of the problems identified above, and include:

(1) developing a database of measured concentrations of target elements and radionuclides in different solid and aqueous materials, from different geographical areas, showing the spatial variability in these concentrations from the regional to the site scale;

(2) developing a database of estimated fluxes of the target elements and radionuclides between different subsurface and surface compartments, showing the spatial and numerical variations in natural fluxes from the regional to the site scale; and

(3) evaluating the measured concentrations and estimated fluxes to construct 'measures' or 'ranges' that can be independently assessed to be safe and against which the outcomes of performance assessments can be evaluated.

Nine countries are participating in the CRP: Argentina, Brazil, China, Cuba, Czech Republic, Finland, Japan, Sweden and the United Kingdom. The countries are undertaking a wide range of individual projects, some of which are based on literature surveys to obtain the required data and others are undertaking primary field work, involving new sampling and analysis. The CRP was effectively launched at the first research co-ordination meeting (RCM) held in Vienna in April 2000 and is planned to run until 2003.

Some of the countries involved in the CRP have previously undertaken independent projects on natural safety indicators, particularly Sweden, Finland, Japan and the UK. Of particular note is the recently completed Japanese PA (known as H12) for a generic HLW repository [JNC, 2000], which included a section which compared the calculated repository release activity concentrations with measured concentrations of naturally-occurring radionuclides in river and groundwaters. This represents the first significant application of natural safety indicators to a major published PA.

Another major ongoing project is the EC sponsored SPIN (testing of Safety and Performance INdicators) project which is aimed at exploring the applicability of various PA output variables (concentrations, fluxes etc.) as performance and safety indicators. This project was launched in the Autumn of 2000 and no results are yet available.

5. FUTURE WORK

Completed and ongoing projects (including the CRP) are encouraging and all point to the fact that that the natural safety indicators method appears to be workable and that it may be a valuable component of a safety case, complementing the dose and risk presentations. This is clearly indicated by the successful application of the method in the Japanese H12 PA.

Further work is necessary to develop the natural safety indicators method to a level where it may be confidently applied in further PAs in a consistent and methodical manner. In particular, work is required to:

— improve databases of natural radionuclide concentrations and fluxes against which to compare repository releases;
— develop appropriate and internationally agreed hazard indices to judge safety in a quantitative manner;
— develop suitable presentational (including graphical) representations of repository-to-nature comparisons that are comprehensible by a wide range of audiences; and
— incorporate natural safety indicators with other indicators (including dose and risk) in a comprehensive and internally consistent safety case for presentation to all stakeholders.

6. CONCLUSIONS

The safety of a radioactive waste repository is normally evaluated in a PA using dose and risk as assessment end-points. Dose and risk are not, however, ideal measures of repository safety because they require assumptions to be made for future human behaviour and environmental change, and their radiological concepts and mathematical treatments are difficult to understand by non-technical audiences. As a consequence, there has been growing interest in using other safety indicators to complement dose and risk, and to provide an independent method for evaluating safety.

One proposed assessment method is to use the distribution of naturally-occurring chemical species in the environment as a datum against which the repository releases can be compared and assessed: the natural safety indicators methodology. Such comparisons can be valid because both the natural and repository species would occur in the same system and their transport behaviour would be controlled by exactly the same processes at the same rates. Comparisons between the natural and repository species may be made in terms of concentration or flux, by mass or by the activity of individual nuclides or by total activity. Although simple in concept, however, there is currently no consensus on the most appropriate comparisons to make or on the interpretation of such comparisons.

A number of recent and ongoing projects have indicated that the natural safety indicators method may be workable and that it may be a valuable component to a safety case, to complement the dose and risk presentations. Further work is necessary, largely to improve databases of natural radionuclide concentrations and fluxes against which to compare repository releases, and to develop appropriate and internationally agreed hazard indices to judge safety in a quantitative manner, and presentational methods.

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SAFETY INDICATORS ADOPTED IN THE FINNISH REGULATIONS FOR LONG-TERM SAFETY OF SPENT FUEL DISPOSAL

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Abstract. A regulatory guide for the safety of spent fuel disposal has recently been issued to guide the implementer's programme in the preconstruction phase. The guide is based on dose criteria in the time frame, which is reasonably predictable with respect to assessment of human exposure. For the time frame that involves major climate changes such as permafrost and glaciation, the guide defines constraints for the activity releases to the environment. This paper discusses the rationale for the selected approach and the derivation of the activity release constraints.

1. RADIATION PROTECTION CRITERIA

The Finnish spent fuel disposal programme has recently passed the decision-in-principle process that is crucial to the selection of the disposal site and to the political acceptance of the disposal plan. The regulator (STUK) participated in the process by reviewing implementer's (Posiva) safety case. The review was based on the general safety regulation issued by the Government in 1999. STUK has recently issued a guide for the long-term safety specifying the general safety regulation.

The Government's general safety regulation includes the following the radiation protection criteria:

In an assessment period that is adequately predictable with respect to assessments of human exposure but that shall be extended to at least several thousands of years:

(1) the annual effective dose to the most exposed members of the public shall remain below 0.1 mSv; and

(2) the average annual effective doses to other members of the public shall remain insignificantly low.

Beyond the assessment period referred to above, the average quantities of radioactive substances over long time periods, released from the disposed waste and migrated to the environment, shall remain below the nuclide specific constraints defined by the Radiation and Nuclear Safety Authority. These constraints shall be defined so that:

(1) at their maximum, the radiation impacts arising from disposal can be comparable to those arising from natural radioactive substances; and

(2) on a large scale, the radiation impacts remain insignificantly low.

Thus, different radiation protection criteria are applied depending on the time frame (environmentally predictable future vs very long-term future) and the spatial extent of impacts (local vs large scale).

In the STUK guide, the radiation protection criteria are clarified as follows:

The dose constraints apply to radiation exposure of members of the public as a consequence of expected evolution scenarios and which are reasonably predictable with regard to the changes in the environment. Humans are assumed to be exposed to radioactive substances released from the repository, transported to near-surface groundwater bodies and further to *watercourses above ground. At least the following potential exposure pathways shall be considered:*

- *use of contaminated water as household water;*
- *use contaminated water for irrigation of plants and for watering animals;*
- *use of contaminated watercourses and relictions.*

Changes in the environment to be considered in applying the dose constraints include at least those arising from land uplift. The climate type as well as the human habits, nutritional needs and metabolism can be assumed to be similar to the current ones.

The constraint for the most exposed individuals, effective dose of 0,1 mSv per year, applies to a self-sustaining family or small village community living in the vicinity of the disposal site, where the highest radiation exposure arises through the pathways discussed above. In the environs of the community, a small lake and a shallow water well is assumed to exist.

In addition, assessment of safety shall address the average effective annual doses to larger groups of people, who are living at a regional lake or at a coastal site and are exposed to the radioactive substances transported into these watercourses. The acceptability of these doses depend on the number of exposed people, but they shall not be more than one hundredth – one tenth of the constraint for the most exposed individuals.

The nuclide specific constraints for the activity releases to the environment are as follows:

- *0,03 GBq/a for the long-lived, alpha emitting radium, thorium, protactinium, plutonium, americium and curium isotopes;*
- *0,1 GBq/a for the nuclides Se-79, I-129 and Np-237;*
- *0,3 GBq/a for the nuclides C-14, Cl-36 and Cs-135 and for the long-lived uranium isotopes;*
- *1 GBq/a for Nb-94 and Sn-126;*
- *3 GBq/a for the nuclide Tc-99;*
- *10 GBq/a for the nuclide Zr-93;*
- *30 GBq/a for the nuclide Ni-59;*
- *100 GBq/a for the nuclides Pd-107 and Sm-151.*

These constraints apply to activity releases which arise from the expected evolution scenarios and which may enter the environment not until after several thousands of years. These activity releases can be averaged over 1000 years at the most. The sum of the ratios between the nuclide specific activity releases and the respective constraints shall be less than one.

The flux of radionuclides from geosphere to biosphere (the geo-bio flux) was selected as the very long-term safety indicator in order to avoid the large uncertainties related to the evolution of the environment due to major climate changes, such as permafrost and glaciation. Consideration of such changes would lead to a wide spectrum of potential biosphere scenarios with very uncertain parameter values. Then the judgement of the safety case might be focussed on overly conservative and extreme bioscenarios.

Nevertheless, the rulemaker must consider various bioscenarios in the derivation of the activity release constraints, as discussed below.

2. DERIVATION OF THE GEO-BIO FLUX CONSTRAINTS

The general safety regulation for long-term safety refers to both local and large-scale radiological impacts. Consequently, both were taken into account in the derivation of the geo-bio flux constraints. As it was anticipated that the local impacts are more limiting, the constraints were initially derived from them. The focus was put on scenarios which are fairly likely, may cause long-lasting radiation exposure of humans having customary living style (i.e. extreme exposures were excluded). The following scenarios were considered:

— Use of a shallow well; household water, garden irrigation and domestic animal watering
— Use of a small lake; fishing, irrigation and grazing at shore
— Use of the sediment of a drained lake; agriculture and soil improvement.

Determination of some parameter values was problematic, in particular the dilution factor for the well scenario. A wide range for this parameter is found in relevant performance assessments:

— The Finnish PA TILA-99: 100 000 m^3/a
— The Swedish PA SITE-94: 10 000 m^3/a
— The Swedish PA SR-97: 2 600 m^3/a.

This very crucial parameter value was determined on the basis on a site specific (the Olkiluoto site) groundwater flow analysis [1]. A number of shallow wells were assumed to exist in the discharge areas of deep groundwater and the proportion of groundwater flow from the repository area to the well was calculated. The analysis resulted in a range of dilution factors from 29 000 to 460 000 m^3/a. On the basis of the recommendation by the authors of the analysis, 90 000 m^3/a was chosen as the dilution factor for the well scenario.

By means of the scenarios given above, dose conversion factors for various nuclides were derived, mainly based on analyses included in references [2] and [3]. These factors and the most critical exposure pathways for some nuclides are as follows:

Nuclide	Dose conversion factor (Sv/Bq)	Critical scenarios/pathways
C-14	5 E-13	Lake/fish and sediment/crop
Cl-36	2 E-14	Well/drinking water and lake/fish
Ni-59	4 E-15	Sediment/crop
Se-79	1 E-13	Lake/fish and well/vegetables
Zr-93	1 E-14	Well/drinking water
Nb-94	6 E-14	Well/external radiation
Tc-99	3 E-14	Sediment/crop
Pd-107	5 E-16	Well/drinking water and vegetables
Sn-126	1 E-13	Lake/fish and well/drinking water
I-129	2 E-12	Well/drinking water and vegetables
Cs-135	3 E-13	Lake/fish and sediment/crop
Ra-226	3 E-12	Well/drinking water and vegetables
Th-229	8 E-12	Well, inhalation and drinking water
Pa-231	1 E-11	Well/inhalation and drinking water
U-238	4 E-13	Well/drinking water
Np-237	1 E-12	Well/drinking water
Pu-239	4 E-12	Well/inhalation and drinking water
Am-243	3 E-12	Well/drinking water and inhalation
Cm-245	3 E-12	Well/drinking water and inhalation

Some consideration was devoted to other scenarios, such as use of a peat bog, but the uncertainties were deemed to be excessive for a rigorous quantitative analysis. However, as clorine and selenium in perticular were found to be enriched in peat, the dose conversion factors of Cl-36 and Se-79 were raised by about one order of magnitude.

The calculated dose conversion factors were adopted for the derivation of the geo-bio flux constraints. The derivation was based on the criteria given in our general safety regulation; it states that the radionuclides originating from the repository may at most result in radiation impacts that are comparable to those arising from natural radionuclides. In Finland, the average external dose from terrestrial radionuclides is 0,5 mSv/a and the average internal dose from natural radionuclides in human body (excluding radon) is 0,3 mSv/a. An individual dose of 0,1 mSv/a was, however, adopted as the reference level in order to leave some margin to more pessimistic scenarios and parameter values than the selected ones.

The geo-bio flux constraints were calculated by dividing the dose rate 0,1 mSv/a and the nuclide specific dose conversion factors. After some further consideration of relevance of various scenarios and roundening of figures, the constraints given in STUK guide (see Chapter 1) were arrived at.

The next step was to check the validity on the constraints with regard to large-scale impacts. Because the proposed site, Olkiluoto, is a coastal site, a seafish consumption scenario is one, which might expose a great number of people. An analysis indicated that the derived geo-bio flux constraints would result in insignificant average individual doses (clearly less than 1 microSv/a) through that scenario.

Another point of comparison was the quantities of natural radionuclides carried to the sea by nearby rivers. There is a regional river, with flow rate a few hundred million m^3/a, the mouth of which is located next to the Olkiluoto island. It can be estimated that through this river, more than 1 GBq/a of long lived natural alpha emitters, 0,1–1 GBq/a of carbon-14 and 1–10 GBq/a of potassium-40 is carried to the potential recipient area of the proposed repository. Some 35 km north of Olkiluoto exists a larger river with flow rate (and natural radionuclide fluxes) about 20-fold in comparison with the former river. It was deemed that the proposed geo-bio flux constraints are small in comparison with quantities of natural radionuclides carried by river to nearby sea bays.

Some consideration was also devoted to releases to the environment of natural long-lived radionuclides (mainly uranium, thorium and radium) from various human practices, such as mining, processing of raw materials and agriculture. It was estimated that such releases from a single source may amount to a few GBq/a, i.e. exceeding clearly the proposed constraints for the same nuclides.

3. DISCUSSION

The release rate of radionuclides from the repository to the environment, the geo-bio flux, was selected as the safety indicator for assessment periods beyond several thousands of years. The main reason for this choice was to exclude from the safety case the great uncertainties related to the evolution of the biosphere in the far future. It would be difficult for the implementer to defend the conservatism (as required in our regulations) of any bioscenario and, as a consequence, the safety case might be based on extreme bioscenarios and overly pessimistic assumptions.

This approach means that the burden on consideration of uncertainties related to evolution of the biosphere in the very long term rests on the rulemaker. He must consider what is a reasonable bioscenario when preparing his regulations, in particular when deriving the geo-bio flux constraints. The implementer need not consider very far future bioscenarios when preparing his safety case.

The derivation of the geo-bio flux constrains was based on both local and large-scale radiation impacts, but in practice, the local impacts were more limiting ones. The local impacts were described by means of a few typical reference scenarios. Consideration of large-scale impacts included e.g. comparisons with fluxes of natural radionuclides via rivers in the vicinity of the proposed disposal site. These comparisons were quite approximate and provided rather further confidence in appropriateness of the geo-bio flux constraints than a real benchmark for their derivation. Hopefully IAEA's CRP on Safety Indicators will bring forth more accurate yardsticks for such comparisons.

Our exercise shows that the geo-bio flux is a practicable indicator of very long-term safety. It has certain pros and cons in comparison with individual dose as a safety indicator. The geo-bio flux is an indicator of the overall containment capability of the disposal system while a dose indicator might focus too much attention to less important peak releases and extreme doses. The performance of most barriers is indicated in units (fractional release rate) which can easily be compared with the ratio of the geo-bio constraint and the maximum inventory of the respective nuclide. On the other hand, a dose constraint incorporates the impacts from all nuclides into one figure while geo-bio flux constraints are nuclide specific. Dose has established acceptability levels whereas there are only few examples of how to benchmark the geo-bio flux constraints.

It should be noted that geo-bio flux constraints and dose constraints are closely bound together via reference biosphere scenarios. One can derive one from the another by means of the dose conversion factors obtained from such scenarios. Thus, development of internationally agreed reference biosphere scenarios would be desirable with regard to the future updating of our regulation.

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THE ROLE OF SAFETY INDICATORS IN GERMANY: THE PAST AND THE FUTURE

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Abstract. As a contribution to the IAEA's "Specialists' Meeting to Resolve Issues Related to the Preparation of Safety Standards on the Geological Disposal of Radioactive Waste" on 18–22 June 2001, an overview of the use of indicators for the demonstration of disposal safety in Germany is given. This includes a description of the present regulatory situation, the utilisation of safety indicators in past assessments, and some ideas concerning the development of new regulations.

1. INTRODUCTION

As part of the demonstration of the post-closure safety of repository systems for radioactive waste the behaviour of complex systems over very long time frames has to be analysed. Properties of the system like its safety, the isolation potential of the system or a subsystem or favourable properties of a potential repository site cannot directly be measured or proven. Therefore, the utilisation of entities which have the potential to demonstrate these properties indirectly – so-called *indicators* – is necessary. The paper presented here gives an overview about how such indicators have been, are and will be used in Germany.

2. CURRENT REGULATIONS IN GERMANY

The German regulatory framework concerning the safe disposal of radioactive waste is based on the Atomic Energy Act ("AtG") of 1994 [1] as well as on additional laws (e.g. mining law) and on the Radiological Protection Ordinance ("StrlSchV") from 1994 [2]. In accordance with the EURATOM guidelines, the latter is currently being revised. In order to provide adequate protection of man and the environment, the Safety Criteria for underground disposal [3] define the individual dose as the main safety indicator for the post-closure phase. A safety analysis based on models for accident scenarios has to show that an individual dose limit of 0.3 mSv/a will not be exceeded. In addition, it has to be shown that the site under consideration has favourable mechanical, tectonical and hydrogeological properties. In 1988, the Reactor Safety Commission (RSK) and the Radiological Protection Commission (SSK), which advise the government in technical questions concerning reactor and waste management safety, recommended to use the dose criterion for time frames up to 10,000 years but to utilise qualitative arguments for considerations going beyond this time [4]. Currently, a revision of the safety criteria for the disposal of radioactive waste is being developed.

3. USE OF SAFETY INDICATORS IN GERMAN ASSESSMENTS

3.1. The Konrad assessment

It is planned to dispose of LLW and ILW in the former iron ore mine Konrad in the Federal State (Bundesland) of Lower Saxony (Niedersachsen). The licensing procedure for the Konrad repository started in 1982 and is still continuing.

Because of the presence of a thick clay cover above the planned repository, groundwater movements at the site are very slow. The geosphere is regarded as the most significant barrier. According to regulation, the individual dose has been used as the major safety indicator in the post-closure safety assessment. No time cut-off has been postulated in these assessments.

The measured groundwater ages of more than $10^{7\ \text{years}}$ indicate groundwater velocities of less than 10^{-5}m/a. This and the salt concentration increase with depth has been used as additional indicators to demonstrate the safety features of the site as well as to support the models used in the assessment.

3.2. The Gorleben site

The Gorleben site in Lower Saxony was intended for HLW disposal in a salt dome. During the last decades, intensive and extensive exploration work has been done at Gorleben. The exploration programme has now been interrupted for at least three and at the most ten years. This time is to be used to resolve a couple of principal questions related to HLW and spent fuel disposal in general and to the suitability of rock salt as a host rock in particular. Since no license for Gorleben has been applied for, only preliminary safety assessments have been carried out in the past. The normal evolution scenario for the post-closure safety assessment assumes that the repository vaults will be closed by creeping salt and that therefore no radionuclide release will take place.

In [5] and [6], a collection of supporting arguments for a possible safety case for Gorleben is given. The behaviour of the system during the past is used as an indicator of its isolation potential in the future. It is argued that the rock salt formation at Gorleben is about 250 million years old. During the time of its existence, there is evidence that no or almost no contact with water from the surrounding formations or the overburden took place. No significant changes of the composition took place, which can be shown using analyses of brine inclusions and bromide concentration profiles.

4. IDEAS AND PROPOSALS FOR FUTURE REGULATIONS

Presently, a revision of the safety criteria for radioactive waste disposal is being developed in Germany. GRS Köln supports the regulator by giving technical advice related to this problem. The ideas given hereinafter reflect the present state of discussion and especially the opinion of GRS Köln.

The concepts developed by the ICRP should be the basis for future regulations. These regulations should respect the following protection principles:

The radiation exposure to be expected from a repository should not be higher than the average variation of natural exposure.

Potential consequences for future generations should not be more serious than consequences accepted today.

Potential consequences abroad should not be more serious than consequences in Germany.

In order to provide the necessary degree of protection, isolation, retention and to some degree dilution of radionuclides should be achieved, based on the principle of multiple barriers. Furthermore, natural systems should be disturbed as little as possible.

During the site selection phase, the following criteria, indicators and properties should be taken into account:

— stability, volume and depth of potential host formations,
— low probability of pathways,

— long transport times,
— geochemical conditions,
— tectonic conditions,
— seismic conditions,
— ability to predict geological evolution,
— absence of natural resources.

The post-closure safety assessment will not be a prediction of the future but an analysis of possible scenarios in order to demonstrate isolation and safety. Risk, dose, concentration and mass flux have the potential to indicate safety. From these indicators, dose should be used as a major indicator. It is intended to treat scenarios separately in the analyses. No integrated (total) risk for the manifold of scenarios under consideration should be calculated. The use of the dose indicator should depend on the likelihood of occurrence for the scenarios. For likely scenarios, 0.3 mSv/a should apply as a dose limit. For less likely scenarios, the value of 0.3 mSv/a should be regarded as a target. This would lead to a "soft-risk approach" where likelihoods of occurrence are taken into account without quantifying them explicitly. Unlikely scenarios (e.g. meteorite impact, extreme changes of site conditions) should not be considered in the analyses because they should be avoided by taking appropriate measures during site selection and design.

The use of safety indicators should also depend on the time frame: For times less than 10,000 years, the value of 0.3 mSv/a should apply as a dose limit for the quantitative analysis. For times beyond that value, it should be seen as a target while additional indicators become increasingly important. For times greater than 10^6years, qualitative safety arguments should be utilised.

While the need for the use of indicators additional to dose is recognised, additional safety indicators should not be prescribed by regulation because they are strongly dependent on the site and the design under consideration. Possibilities for the demonstration the isolation of the system and of subsystems are low groundwater fluxes, velocities, mass fluxes and low fractions of possibly released material, a high retardation as well as long transport times. Indicators and arguments supporting the analysis and enhancing confidence are the absence of groundwater flux, high groundwater ages, and/or reducing chemical conditions. The value of such additional indicators also lies in their potential to improve communication with non-technical audiences: e.g. risk might be better understood than dose because it can be compared to other risks occurring in life. The comparison of fluxes might illustrate the isolation potential of certain barriers.

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APPLICATION OF NON-DOSE/RISK INDICATORS FOR CONFIDENCE-BUILDING IN THE H12 SAFETY ASSESSMENT

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Abstract. In the H12 study, non-dose/risk safety indicators have also been considered with a view to increasing confidence in the safety assessment. The H12 safety assessment considers system evolution for a range of scenarios; not only normal groundwater scenarios but also isolation failure scenarios due to unlikely natural disruptive events. The calculated nuclide concentrations and fluxes in the surface environment for the reference groundwater scenario were compared with measurements of naturally occurring nuclides. This comparison indicated that the concentration and fluxes of radionuclides released from the repository would be several orders of magnitude lower than those of natural radionuclides. There may exist cases, such as some natural disruptive events, where the likelihood of occurrence is extremely low and the "Reference Biosphere" approach is difficult to be applied for biosphere modelling. The use of qualitative assessment to allow comparison with naturally occurring nuclides based on observations of natural systems may play a role in supporting the robustness of the system concept. These examples suggest that relevant application of these non-dose/risk indicators supports a more robust case. An advantage to applying such indicators is that both technical and non-technical audiences can judge the relative, long-term impact of a deep geological repository.

1. INTRODUCTION

The second progress report, H12 (e.g. [1,2]), which is aimed at establishing a technical basis for HLW disposal in Japan, represents an important milestone in the development of the Japanese HLW program [3]. The concept of geological disposal in Japan is similar to that being considered in other countries, being based on a system of multiple passive barriers consisting of the geological environment (geosphere) and an engineered barrier system (EBS; including vitrified waste, an overpack and buffer material). The disposal system considered is generic, in the sense that no host rocks or siting areas have so far been identified.

Geological disposal systems can be 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 minimized through the use of passive barriers with multiple safety functions.

The safety of a geological disposal system can be assessed using defined indicators such as risk and dose, which directly measure radiological effects on human beings. The only scenarios modeled in detail in the H12 safety assessment for dose calculation were groundwater scenarios (scenarios in which moving groundwater provides the pathways for transport of radionuclides from the repository to the surface environment). These included:

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— a Base Scenario, in which external events and processes such as natural geological and climatic phenomena, initial defects and future human activities are excluded, and
— a set of perturbation scenarios, in which the potential impacts of external events and processes are examined.

A Reference Case was defined for the Base Scenario, incorporating a particular set of design features, model assumptions and parameter values. Alternative cases were also defined for the Base Scenario (with alternative design features, model assumptions or parameter values) and for the perturbation scenarios.

Since the safety assessment of a geological disposal system covers a very long period of time, there are inherent uncertainties involved in assumptions about future human activities and environmental conditions, which increase the longer the time horizon becomes. In order to address some of these uncertainties, Reference Biospheres (e.g. [4]) have already been adopted in various performance assessments overseas and this concept is used in the main body of the H12 safety assessment in order to illustrate the distribution of radionuclides in the biosphere and resulting radiological consequences over timescales of interest.

To further allow for uncertainties, it has also been proposed to supplement assessment results based on dose (or equivalent risks) by other independent indicators, to avoid relying on assumptions made in the biosphere model. The use of non-dose/risk indicators, such as concentration and flux of specific nuclides and radiotoxicity index, is useful as they are more indicative of the isolation capability of a disposal system and the potential risks associated with radioactive waste ([5–7]).

On the other hand, FEPs that could generate “isolation failure scenarios” (scenarios in which the human environment is affected due to the physical isolation of the waste being compromised), for example associated with natural disruptive events, were screened out (e.g. on the basis that they could be excluded or reduced to a very low probability by siting). Nevertheless, some less formal "what if?" analyses using non-dose/risk indicators have been carried out to illustrate the magnitude of potential consequences from disruptions, and thus the importance of siting the repository in a suitable environment.

This paper illustrates how non-dose/risk indicators have been applied to supplementary assessments for the Reference Case, and more qualitative “what if?” analyses for natural disruptive events, to increase confidence in the H12 safety assessment.

2. REFERENCE CASE

The Reference Case was defined as a baseline for the calculations in the H12 safety assessment. The following assumptions define a starting-point for investigating the Reference Case:

— The current conditions of the geological environment remain unchanged indefinitely;
— The EBS functions as expected;
— The current conditions of the surface environment remain unchanged indefinitely.

All radionuclides released from the vitrified waste packages in the repository are assumed to migrate in channels in undisturbed host rock and then upwards through major water-conducting faults (MWCF) to a shallow aquifer which, in turn, discharges to a river. Significant dilution occurs as a small amount of groundwater from the aquifer enters the river.

In the performance assessment retardation processes and radionuclide travel times are considered in the fault but are conservatively ignored for the aquifer. The nuclide transport pathways considered in the Reference Case are illustrated in Figure 1.

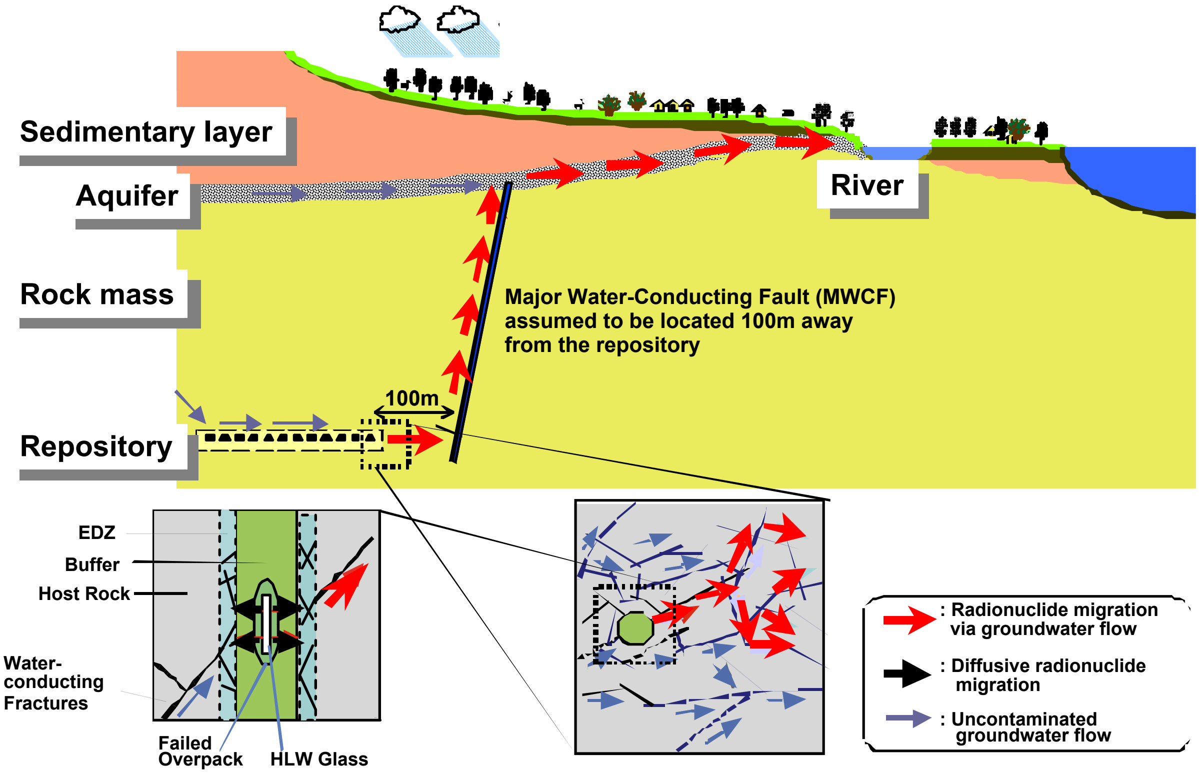


FIG. 1. Nuclide transport pathways considered in the Reference Case.

Release to the biosphere was assumed to be via a MWCF (retardation taken into account), discharging into a shallow aquifer (no retardation considered) and then into a river (flow rate 10^8 m^3 y^{-1}). This is a reasonable choice since, in Japan, the majority of drinking water (70%) and irrigation water (85%) is taken from surface waters such as rivers and dams.

The biosphere model, based on the Reference Biosphere concept, 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 through which radionuclides could enter the food chain, along with uptake and concentration factors, are also defined [8]. The resulting dose (from ingestion, inhalation 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.

The basic model chain mentioned above was used to perform calculations for a repository containing 40,000 packages of vitrified waste. Figure 2 shows the calculated results for the Reference Case as a function of time after repository closure. In the figure, the thick line shows summed dose from all the safety-relevant radionuclides and the other lines show the largest contribution to the dose at different times from particular nuclides. Figure 2 also indicates the periods for which key assumptions regarding components of the disposal system are likely to remain valid as shaded areas, in order to facilitate understanding of these results.

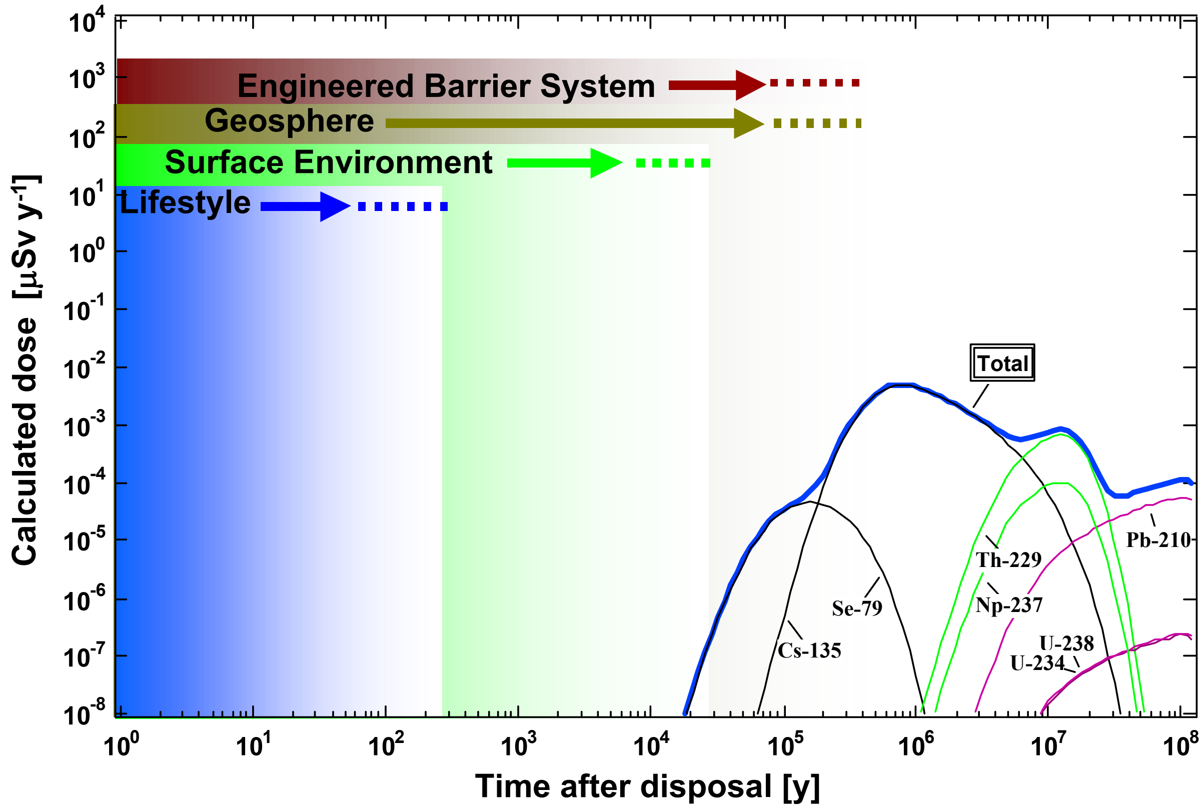


FIG. 2. Reference Case dose evaluation.

(Shaded areas indicate the period for which modeling of each system component can be considered reliable.)

3. SUPPLEMENTARY ASSESSMENT FOR THE REFERENCE CASE

3.1. Effectiveness of barrier performance

Indicators which show the effectiveness of barrier performance are useful for demonstrating redundancy within the natural and engineered barrier system, i.e. that safety is not dependent on the performance of a single part of the disposal system. Such presentations can be a highly transparent means of communicating a safety concept to both experts and the public, even though these alone are inadequate indicators of total system safety.

Radiotoxicity

The long containment times within the overpack, release from the glass matrix, transport through the bentonite and transport through the rock mean that many radionuclides decay to insignificant levels within these barriers, before reaching the human environment. Figure 3 illustrates this, showing a measure of the total radiotoxicity of the nuclides contained within the different barriers as a function of time, as calculated for the Reference Case. The radiotoxicity index is defined here by the following equation:

Radiotoxicity of nuclide i [m^3] =
Inventory of nuclide i in a specific compartment [Bq]
/ (DWC_i [Bq cm^{-3}] $\times 10^6$ [cm^3 m^{-3}]),

where DWC_i [Bq cm^{-3}] is the maximum permissible concentration of nuclide *i* in drinking water outside the monitored area [9].

The figure indicates that:

— radiotoxicity is entirely contained within the glass matrix until the overpacks are breached at 1000 years following emplacement, by which time it has declined by one order of magnitude due to radioactive decay;
— after 10^4 years, radiotoxicity is still contained predominantly within the glass matrix, and has declined by two orders of magnitude to less than that of 0.01 km^3 of granite with 1ppm uranium concentration;
— 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 10^6 – 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 significant proportion of the radiotoxicity contained within the biosphere.

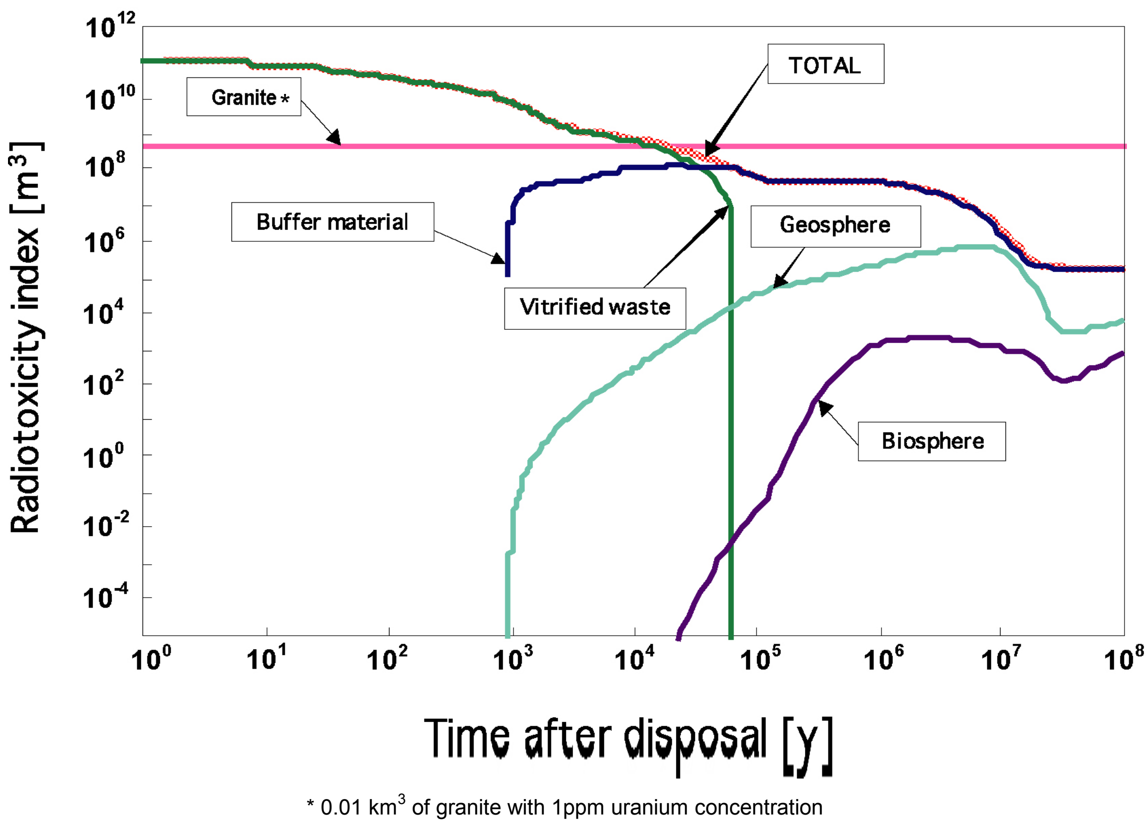


FIG. 3. Radiotoxicity index as a function of time for the Reference Case, showing containment of radiotoxicity in the various components of the disposal system for one waste package.

Radiotoxicity release rate

The time-dependent release rates from the vitrified waste, EBS, host rock and MWCF for the Reference Case are shown in Figure 4. The rates are expressed using a "normalized radiotoxicity release-rate index", defined by the following equations:

Normalized radiotoxicity release-rate index of nuclide *i* [y^{-1}] =
Radiotoxicity release-rate index of nuclide *i* [$m^3\ y^{-1}$]
/ Radiotoxicity index of nuclide *i* in vitrified waste at the time of disposal [m^3],

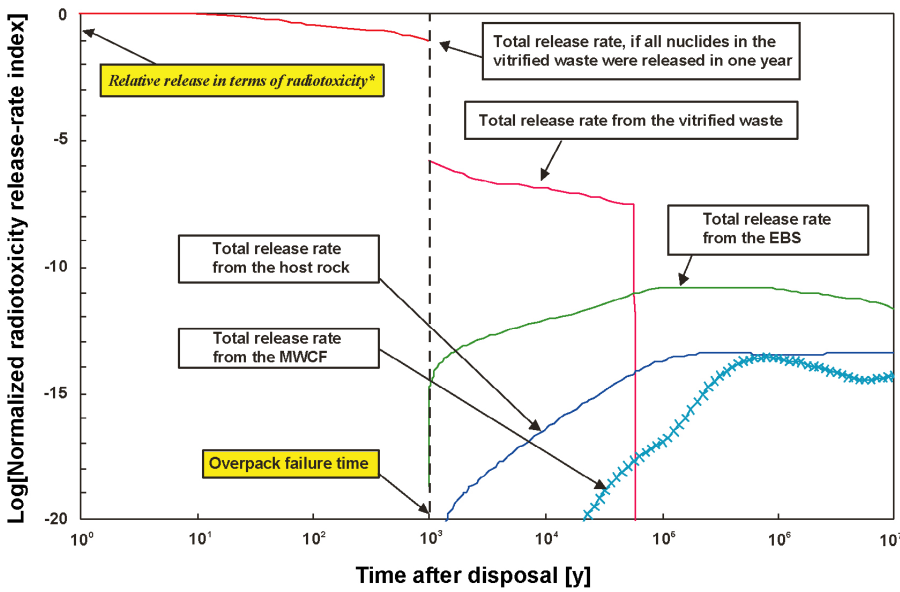
where

Radiotoxicity release-rate index of nuclide *i* [$m^3\ y^{-1}$] =
Release rate of nuclide *i* [$Bq\ y^{-1}$] / (DWC_i [$Bq\ cm^{-3}$] $\times 10^6$ [$cm^3\ m^{-3}$]),

and

Radiotoxicity index of nuclide *i* in vitrified waste at the time of disposal [m^3] =
Inventory of nuclide *i* in vitrified waste at the time of disposal [Bq]
/ (DWC_i [$Bq\ cm^{-3}$] $\times 10^6$ [$cm^3\ m^{-3}$]).

Figure 4 shows that, in the Reference Case, all the barriers contribute to the low biosphere releases. The release of radiotoxicity from the vitrified waste, which finishes abruptly when the glass is completely dissolved, is 5 to 7 orders of magnitude smaller than the total release rate, if all nuclides in the vitrified were released in one year at 1,000 years after disposal, indicating the effectiveness of the release control due to the slow dissolution of the glass waste form. In particular, the release rate from the EBS to the host rock is 3 orders of magnitude smaller than that from the vitrified waste. A further reduction of 2 to 3 orders of magnitude is attributable to transport through the host rock, with an additional small reduction due to transport through MWCF.



* Unity corresponds to the release of the total radiotoxicity in the vitrified waste at the time of disposal in one year.

FIG. 4. Comparison of release rates from each barrier component using radiotoxicity as an index.

Thus, both the EBS and the natural barrier contribute significantly to the reduction of radiotoxicity released to the biosphere. In addition, Figure 4 illustrates the delays in the time to the peak of the releases from the outer or “downstream” barriers. These, in combination, demonstrate the allocation of performance among individual components of the multibarrier system.

3.2. Concentration and fluxes in the environment

Safety indicators such as nuclide concentration and flux can usefully supplement dose calculations in the evaluation of overall system performance.

Concentration of natural radioactivity

Figure 5 shows a comparison of the calculated concentrations of 4N+2 (U-238) chain members released to a river for the Reference Case with the range of naturally occurring concentrations of these nuclides [10–13], as well as the maximum permissible uranium concentration, according to the Drinking Water Quality Guidelines of the World Health Organization [14]. The comparison shows that the concentrations of radionuclides released from a repository would be several orders of magnitude lower than that of natural radionuclides. Further discussion is presented in [15].

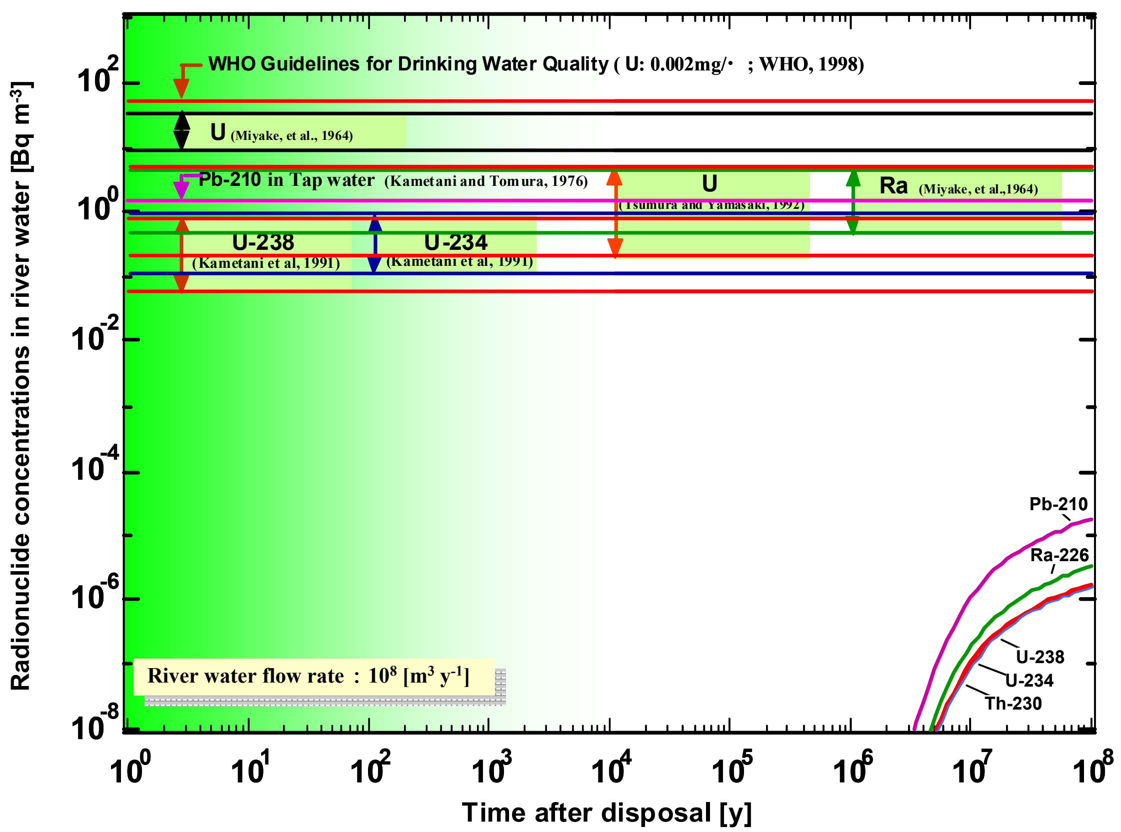


FIG. 5.Comparison of calculation results (Reference Case) and natural radionuclides: concentration of 4N+2 natural decay series nuclides in river water.

Equivalent fluxes of U-238

Direct comparisons such as those above are possible for radionuclides that exist in nature, but arguments based on analogy are required for those nuclides which do not occur naturally. To complement these as proposed in [6], one possibility is comparisons based upon radiotoxicity, e.g., by dividing the concentrations in Bq L^{-1} by the respective values of annual limit of intake (ALI). This has been applied in [15] as "the converted dose" which was calculated due to nuclide concentration in river for the H12 Reference Case converted to dose using ALI [10]. In this paper, the radionuclide flux from the geosphere (i.e. for the MWCF) in the H12 Reference Case can be converted to equivalent fluxes of U-238 using ALI [10]. These results are shown in Figure 6.

It can be considered that there are many types of (surface) water systems that could receive radionuclides released from the geosphere. River water is one of the most likely. In Figure 6, the equivalent flux of U-238 from the Reference Case is compared with that of natural radionuclides fluxes based on measured U-238 and U-234 concentration in 17 rivers of around 10^8 to 10^9 m^3 y^{-1} flow rate [12], which are comparable with the flow rate 10^8 m^3 y^{-1} for the H12 biosphere model in the Reference Case. It is apparent that the equivalent flux due to the repository is several orders of magnitude lower than that of naturally occurring uranium. It is still lower than the range of natural radionuclides fluxes of groundwater based on measured U and Th concentration in groundwater [16] and an average pumping rate ($2\times10^6 m^3 y^{-1}$) of deep well water [17].

In the characterization phase at a specific site, it will be possible to measure site-specific concentrations and fluxes of naturally occurring radionuclides and to compare them with the results of performance assessment models.

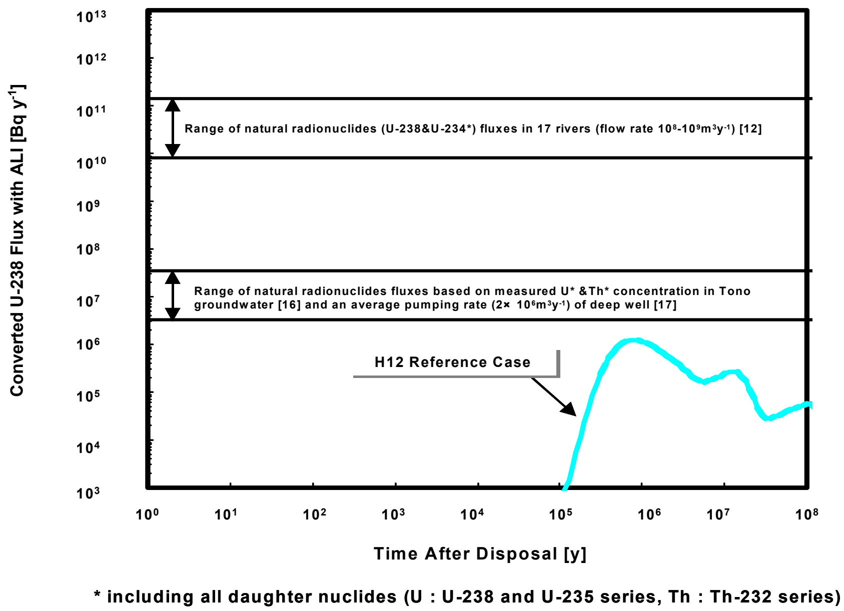


FIG. 6. Assessment of the impact from non-naturally occurring radionuclides in the repository using the converted flux of U-238.

4. QUALITATIVE ASSESSMENT FOR NATURAL DISRUPTIVE EVENTS

There is an extremely low likelihood that some kind of natural disruptive event will impact a well-sited repository directly. For "what if?" calculations, these events are nevertheless assumed to occur at about 10^5~10^6 years after disposal, when the remaining radiotoxicity of the repository is comparable with that of uranium ores or a relevant volume of host rock as shown in Figure 3.

For isolation failure scenarios due to natural disruptive events, such as uplift and erosion and volcanic activity, when the repository is exposed at, or mobilized to, the ground surface, doses can be estimated by assuming that all the radionuclides released (mainly in solid form) enter river water. Under these extreme conditions, however, the application of the Reference Biosphere approach, which is subject to a number of uncertainties, becomes more difficult. As discussed in the previous section, fluxes in groundwater and surface waters can be compared with fluxes from the repository into the environment for groundwater scenarios. However, these solute fluxes are difficult to relate to fluxes mainly in solid form due to natural disruptive events. Hence, the consequences of these extreme scenarios are estimated in terms of radionuclide fluxes in solid form and these are compared with those of naturally occurring radionuclides to allow a more robust assessment of consequences.

4.1. Uplift/erosion

Based on estimations from topographic and geological data, it appears that most parts of Japan have been subject to some uplift over the last million years, but the average uplift rate is generally less than 100 m per 100,000 years (1mm per year) if consideration is given only to areas outside mountainous regions. Evidence of significant erosion by glaciers can be seen only at high altitudes in Japan, which supports the assumption that erosion at potential repository sites is mainly caused by weathering, giving an average denudation rate of 10 to 100 m per 100,000 years (0.1~1 mm per year).

The potential effects of uplift and erosion on the safety functions of the geological disposal system are analyzed by assuming that the rate of erosion is equal to that of uplift, since this would result in the greatest consequences. The consequences of an isolation failure scenario, in which the repository actually becomes exposed at the ground surface after entering an oxidizing/higher-flow zone near the ground surface, are analyzed by means of a less formal "what if?" calculation. 1 mm y^{-1} and 0.1 mm y^{-1} uplift rates and pessimistically the same erosion rates are assumed.

It is rather arbitrarily assumed that the groundwater becomes oxidizing and its flow velocity increases by one order of magnitude when the repository depth is reduced to less than one hundred meters (Figure 7). Nuclide transport after this is calculated using solubilities, distribution coefficients and diffusion coefficients for oxidizing conditions, although before this it is based on the Reference Case assumptions.

The fluxes of nuclides resulting from this scenario are estimated from the rate of erosion of a volume of material containing the repository and the average concentration of radionuclides in this volume. The volume has an area equal to that of the repository and a rather arbitrary thickness of 100 meters (the thickness of the oxidizing zone: see Figure 7). The calculated nuclide fluxes are compared to naturally occurring nuclide fluxes, calculated for a rock mass (granite or uranium ore) of the same area containing uranium of various grades, assuming the same erosion rates. For comparison purposes, these nuclide fluxes (U-238, U-235 and Th-232 decay series for granite and U-238 and U-235 decay series for uranium ore) are converted to equivalent fluxes of U-238 using the ALI as described above.

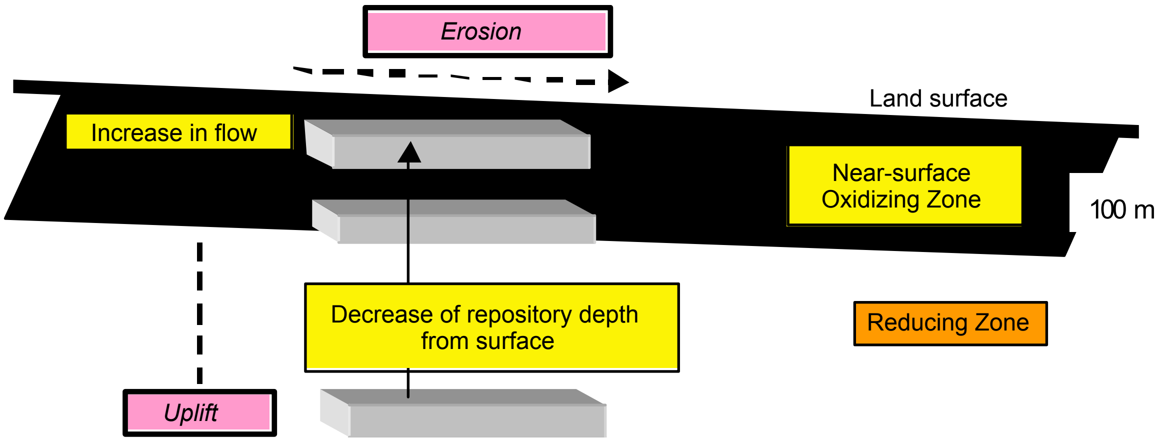


FIG. 7. Conceptual model of uplift/erosion.

Results are calculated for two uplift/erosion rates and two initial repository depths: 1,000 m (crystalline rock and Pre-Neogene sedimentary rock: hard rock) and 500 m (Neogene sedimentary rock: soft rock). The earlier the repository reaches the ground surface, the greater the calculated fluxes (Figure 8). Results are relatively insensitive to initial depth for the faster erosion rate, but more sensitive for the slower rate. This is because, for the slower erosion rate, Np-237 (which is a key nuclide with a half-life of about two million years) begins to decay significantly before the repository is exposed at the surface.

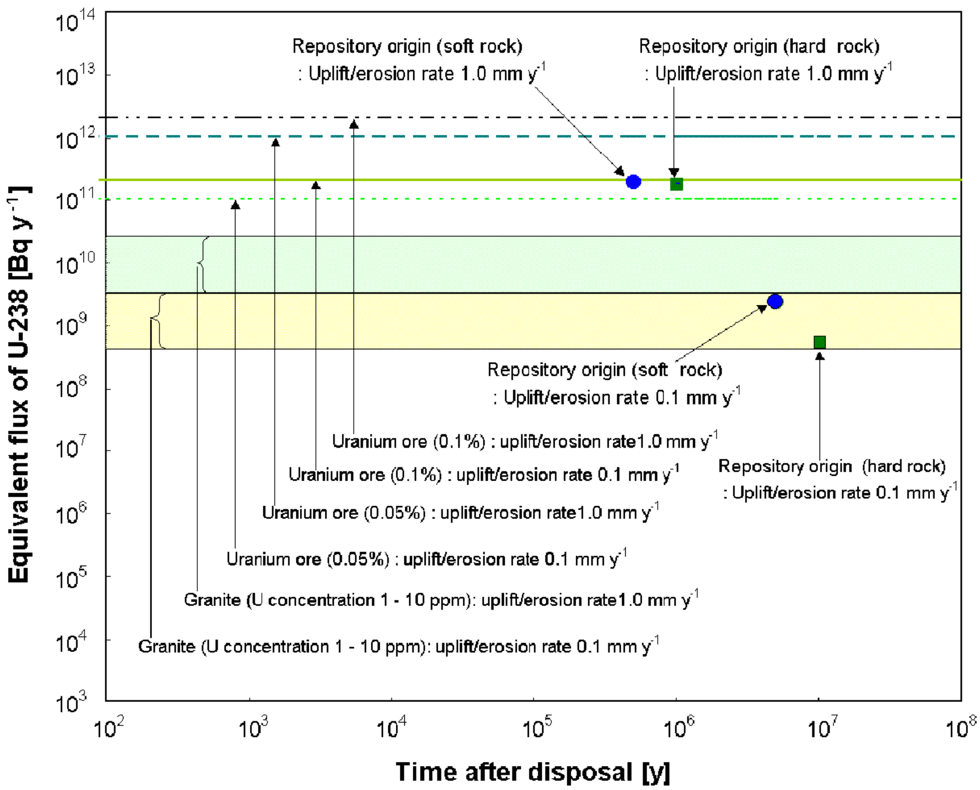


FIG. 8. Comparison of fluxes of nuclides due to the repository with fluxes of natural origin for the uplift/erosion scenario.

4.2. Volcanic activity

Tracing back the history of volcanic activity in the Quaternary in Japan shows that the locations where such activity occurs are restricted to distinct regions and that there is little change in these locations. In addition, the direct effects of volcanic activity which could significantly influence repository performance have been studied and are expected to be restricted to a maximum of a few tens of kilometers from the activity centers. Detailed study of the spatial and temporal patterns of volcanism in Japan can indicate those areas where the probability of new volcanism is extremely low, indicating appropriate areas for further consideration for repository siting.

Volcanic activity could potentially mobilize large amounts of radioactive and toxic materials [18]. For a "what if?" analysis, the following case which assumes that:

- volcanic activity occurs in the repository area (in spite of the very low probability of new volcanoes, as noted above);
- a major magmatic intrusion into the repository takes place

is selected, since it would result in the greatest consequences. In order to evaluate the transport of nuclides to the surface in this case, the additional assumptions are made that:

- the intrusion occurs one hundred thousand years after disposal;
- all the radionuclides confined in the EBS are released.

Expressing releases in terms of an equivalent flux of U-238 as above, the release from one waste package is equivalent to the natural U with 1 ppm concentration (U-238, U-235 and Th-232 series considered) contained in a volume of lava of 4×10^5 m^3. The average volume of a Japanese volcano is 4×10^{10} m^3, although this amount of lava is not necessarily discharged in a single eruption. It can be inferred that nuclide release from the whole repository (40,000 waste packages) by a volcanic eruption would not be significantly greater than that if the lava erupted at the same time.

5. CONCLUSIONS

As dose estimates inevitably include significant uncertainties from biosphere modeling, non-dose/risk indicators have been investigated. This study presents illustrative examples of application of non-dose/risk indicators in the H12 safety assessment.

For groundwater scenarios, concentration and fluxes in groundwater and surface water are useful to supplement dose results for overall system performance. The comparison indicated that the concentration and fluxes of radionuclides released from the repository for the Reference Case would be several orders of magnitude lower than that of natural radionuclides. This suggests that application of non-dose/risk indicators can put the long-term safety of the repository system into perspective.

For isolation failure scenarios due to disruptive natural events such as uplift/erosion and volcanic activity, fluxes in solid material are relevant indicators for comparison with naturally occurring radionuclide fluxes due to these events.

When conducting such a comparison, it is important that actual measurements of natural radionuclide concentration or fluxes exist as a comparable standard, that the definition of the disposal system is clear and that the uncertainties involving long-term safety are considered.

Moreover, for the comparison, the choice of indicators due to naturally occurring radionuclides in groundwater or solid material is needed to treat relevant scenarios. Considering the remaining radiotoxicity of the repository, qualitative assessments using non-dose/risk indicators are relevant for scenarios which are very unlikely and not expected to occur before a certain time period (e.g. 10^5 years).

Relevant application of these non-dose/risk indicators supports a more robust case. An advantage to making such a comparison is that both technical and non-technical audiences can judge the relative, long-term impact of a deep geological repository.

This study was made for a generic disposal system concept, i.e. it targeted neither a particular rock type nor a particular geographical area. In the characterization phase at a specific site, measurements of concentrations and fluxes of naturally occurring radionuclides at a specific site can be compared to the output of performance assessment models.

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