Radioactive Waste Management

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Safety Cases for Deep Geological Disposal of Radioactive Waste: Where Do We Stand?

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The OECD Nuclear Energy Agency (NEA) was established on 1st February 1958 under the name of the OEEC European Nuclear Energy Agency. It received its present designation on 20th April 1972, when Japan became its first non-European full member. NEA membership today consists of 28 OECD member countries: Australia, Austria, Belgium, Canada, the Czech Republic, Denmark, Finland, France, Germany, Greece, Hungary, Iceland, Ireland, Italy, Japan, Luxembourg, Mexico, the Netherlands, Norway, Portugal, Republic of Korea, the Slovak Republic, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The Commission of the European Communities also takes part in the work of the Agency.

The mission of the NEA is:

- to assist its member countries in maintaining and further developing, through international co-operation, the scientific, technological and legal bases required for a safe, environmentally friendly and economical use of nuclear energy for peaceful purposes, as well as
- to provide authoritative assessments and to forge common understandings on key issues, as input to government decisions on nuclear energy policy and to broader OECD policy analyses in areas such as energy and sustainable development.

Specific areas of competence of the NEA include safety and regulation of nuclear activities, radioactive waste management, radiological protection, nuclear science, economic and technical analyses of the nuclear fuel cycle, nuclear law and liability, and public information. The NEA Data Bank provides nuclear data and computer program services for participating countries.

In these and related tasks, the NEA works in close collaboration with the International Atomic Energy Agency in Vienna, with which it has a Co-operation Agreement, as well as with other international organisations in the nuclear field.

Cover credit: Deep Geologic Repository Concept at the Bruce Site (Canada).

FOREWORD

A safety case for the post-closure phase of a geological repository is a synthesis of evidence, analyses and arguments to quantify and substantiate that a repository will be safe after closure and beyond the time when active control of the facility can be relied upon. A safety case is presented, most often by organisations responsible for implementing waste disposal solutions, at specific points in the process of repository development. The International Symposium on Safety Cases for Deep Geological Disposal: Where Do We Stand? (held in January 2007 in Paris, France) covered practical experience in defining, planning and developing the elements of a safety case for deep geological disposal of radioactive waste. Presentations by invited speakers, including stakeholders, also addressed the communication and presentation of safety cases.

The aims of the symposium were to:

- Share practical experiences on preparing for, developing and documenting a safety case both at the technical and managerial levels (testing the concept of a safety case).
- Share experiences on the regulatory perspective. What are the regulatory requirements and expectations of the safety case? Does the safety case provide answers?
- Highlight the progress made in the last decade, the actual state of the art and the observed trends.
- Assess the relevance of the international contributions in this field.
- Receive indications useful to the future work programmes of the NEA and other international organisations.

The symposium was organised by the Nuclear Energy Agency (NEA) of the Organisation for Economic Co-operation and Development (OECD), in co-operation with the European Commission (EC) and the International Atomic Energy Agency (IAEA). Participants in the symposium included approximately 150 representatives from implementer, regulatory, scientific and stakeholder organisations in nearly 20 countries and international organisations.

These symposium proceedings represent a multitude of contributions in the development and communication of the safety case made during the past 15-20 years, thereby building on the experience and knowledge exchanged at the previous international symposium of 1989, also organised by the OECD Nuclear Energy Agency, the EC and the IAEA. In addition to the contributed papers, these proceedings include an overview of the symposium, a synthesis of lessons learnt from an international perspective and detailed session-by-session summaries of the symposium.

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Peter De Preter	Agency for Radioactive Waste and Enriched Fissile Materials, ONDRAF/NIRAS, Belgium
Bruno Cahen	Agency for Radioactive Waste Management, Andra, France
Sylvie Voinis	Agency for Radioactive Waste Management, Andra, France
Gérard Bruno	Radiological Protection and Nuclear Safety Institute, IRSN, France
Hiroyuki Umeki	Japan Atomic Energy Agency, JAEA, Japan
Jesús Alonso	Empresa Nacional de Residuos Radiactivos, SA, ENRESA, Spain
Bo Stromberg	Swedish Nuclear Power Inspectorate, SKI, Sweden
Piet Zuidema	National Cooperative for the Disposal of Radioactive Waste, NAGRA, Switzerland
Alan Hooper	United Kingdom Nirex Limited, United Kingdom
Abraham Van Luik	Office of Civilian Radioactive Waste Management, Department of Energy, DOE, United States of America
Michel Raynal	European Commission, EC, Belgium
Didier Louvat	International Atomic Energy Agency, IAEA, Austria
Claudio Pescatore	Organisation for Economic Co-operation and Development, Nuclear Energy Agency, OECD/NEA, France
Betsy Forinash	Integration Group for the Safety Case, IGSC, Secretariat, OECD/NEA, France

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OVERVIEW OF THE SYMPOSIUM

This overview section highlights the context and aims of the 2007 Symposium on Safety Cases for Deep Geological Disposal of Radioactive Waste: Where Do We Stand? and provides a short overview of the presentations and discussions at the symposium.

Context for the symposium

The Organisation for Economic Co-operation and Development (OECD) Nuclear Energy Agency (NEA), in co-ordination with the International Atomic Energy Agency (IAEA) and the European Commission (EC), sponsored a symposium in 1989 on state-of-the-art assessment of long-term safety of geologic disposal facilities for radioactive wastes. Whether the tools existed to evaluate repository performance over the long term was answered positively at that time. The opportunity was also recognised for international organisations to play a constructive role in bringing together national implementer and regulatory bodies to develop common understanding of several outstanding issues. These issues were addressed by successful specialist workshops, meetings and symposia. Among the issues addressed, and now better appreciated, were the degree to which the concept of validation could be applied to far-future modelling; the methods by which uncertainties could be evaluated, explained and managed; and approaches to present technical material to non-specialist audiences, including societal decision makers.

It was realised that building confidence in the long-term safety of a geologic disposal system requires more than only the results of calculations. In particular, it is necessary to demonstrate that the result of a calculation merits sufficient confidence to support a decision and that the result is compatible with current knowledge. It is also necessary to provide transparent explanations of how a sufficient degree of confidence is achieved in the face of uncertainty. The aggregation of evidence supporting the safety of a geologic disposal system has been termed a safety “case” or “dossier”. This terminology was adopted to underscore that the expertise required to create such a case goes beyond the ability to develop and exercise a mathematical model. While a mathematical safety evaluation lies at the heart of the safety case, it is not the whole of the case.

To address this expanded view of what is required to demonstrate repository safety, three international agencies – those being the IAEA, EC and OECD Nuclear Energy Agency – have sponsored both the 1989 symposium and this 2007 symposium. These three agencies also sponsored work that supported the definition and development of safety cases. During the 2007 symposium, each of these agencies provided an overview of activities. Specifically, the IAEA has developed a new safety standard report for geologic repositories, co-sponsored by the NEA, which includes an explicit definition of a safety case and lists higher-level requirements. The NEA, with its history of projects and activities on the safety case, established the Integration Group for the Safety Case to define and address the safety case concept and its requirements. Products and publications from the Integration Group for the Safety Case provide information and methods to help support technical developments. The European Commission has sponsored numerous technical projects, as well as several on societal aspects, designed to further develop the safety case concept as a practical working framework.

Aims of the symposium

The aims of the symposium were to:

- share practical experiences on preparing for, developing, and documenting a safety case both at the technical and managerial level (test the concept of a safety case);
- share experiences on the regulatory perspective. What are the regulatory requirements and expectations of the safety case? Does the safety case provide answers?
- highlight the progress made in the last decade; the actual state of the art and the observed trends;
- assess the relevance of the international contributions in this field;
- receive indications useful to the future working programme of the NEA and other international organisations.

These aims were addressed over four sessions. The first session addressed the progress of safety cases in the country of France. The second session highlighted the evolution of the safety case concept internationally and the development of the safety case concept in several national programmes that served as case studies (Appendices A and B). The third session focused on practical experience in the development and review of safety cases, including the regulatory perspective (Appendix C). The fourth session examined the actual or potential role of a safety case in societal dialogue (Appendix D). A poster session covered these aspects and more (Appendix E). A fifth session consisted of a discussion of the important outcomes of the symposium. A final session of closing remarks by the sponsoring agencies completed the symposium. A list of the symposium participants is provided as Appendix F.

Symposium summary

Session I (see Appendix A for papers presented in this session) provided an overview of the French programme in terms of legal, regulatory and implementer perspectives, which were co-ordinated to produce a plan for development of a repository and an integrated storage system, to promote and integrate advances in the fuel cycle and to embark on public dialogue in support of each area. This session provided an example of a national programme becoming highly coherent at various levels, including the societal level. All three levels represented in the presentations, government regulator, implementer and the public, made a point that international co-operation was important to the overall effort.

Session II (see Appendix B for papers presented in this session) was divided into two parts. The first part of Session II reviewed the evolution of the safety case concept in the context of relevant work by four key international organisations: the Nuclear Energy Agency, the International Commission on Radiological Protection, the International Atomic Energy Agency and the European Commission. The NEA presentation emphasised the evolution from the safety assessment to the broader concept of the safety case, with additional information and institutional aspects that contribute to confidence in safety. The new safety standard report for repositories, established by the IAEA, was jointly published with the Nuclear Energy Agency in 2006 (*Geological Disposal of Radioactive Waste, Safety Requirements*, Jointly Sponsored by the IAEA and the NEA, IAEA Safety Standards Series No. WS-R-4, International Atomic Energy Agency, Vienna). The definition of a safety case and its requirements in WS-R-4 were fully endorsed by the NEA and are reflected in NEA publications on the safety case. The EC has also sponsored significant work on safety assessments. The EC endeavour has been broadened to support the concept of bringing to bear all information that supports an evaluation of long-term safety: a safety case. Guidelines from the ICPR International Commission on Radiological Protection relate specifically to performance projections in the safety case of potential radiation exposure in the distant future. These and other proposed changes to update and harmonise recommendations were discussed in the first part of Session II.

The second part of Session II addressed the range of safety criteria applied in national programmes for geological disposal, noting not only differences in numerical criteria but also in the scope and timeframes required for safety assessments and the weight assigned to the criteria alongside other factors to be considered. Presentations on two national programmes highlighted the numerous factors that can contribute to the development and evolution of safety cases over time for specific sites and disposal concepts, including, for example, the regulatory environment, advancement of modelling methodologies and availability of more detailed and site-specific data.

Session II concluded with a plenary panel discussion. In particular, two topics held the interest of participants: the nature and definition of the safety case and the variety of regulatory requirements in national programmes for geological disposal.

There was significant discussion of the nature of the safety case in the context of decision making, contrasting a narrower view as the case being made for safety (a “best-foot forward” approach), and a broader perspective that includes full discussion of remaining open issues and work planned to address open issues. Many participants favoured the broader view, whereby a safety case is considered a product to support a decision specific to one phase of repository development but also recognises the information and confidence needs for other phases of the repository development and lifetime. Other participants expressed the view that a safety case may not have “open issues”, but acknowledged that it could put forth assumptions that call for further evidence to be provided in later stages to support confidence in long-term safety. Thus, whatever the terminology applied, as time goes on and more information becomes available, the safety case is refined to support the next decision point.

The plenary discussion also acknowledged the variety of regulatory requirements among national programmes. Recognition was given to the difficulty in harmonising national requirements and consideration was given to reasons for significant differences in approach. A unified view on how to address this issue or whether it should be addressed was not derived. There was consensus, however, that it would be inappropriate simply to compare the numerical outcome of a safety assessment for one national programme to the outcome of another. In any comparison, many issues need to be taken into account such as the inventory being disposed, the way regulations may specify the treatment of exposure pathways and whether the approach to the safety assessment was selected to be bounding and conservative or to aim for a more realistic assessment of expected evolution. The presentations and discussion in this area drew very much on current and ongoing work by the NEA Waste Regulators’ Forum.

Session III (see Appendix C for papers presented in this session) focused on recent experience with building, reviewing and planning safety cases. Session III was divided into three sub-sessions. In the first sub-session, presentations suggested that not all aspects of the safety case are necessarily defined in regulations; this appears especially true for more qualitative or institutional aspects. The implementer, however, is well served by adopting a safety case approach, realising that this approach results in a more comprehensive case for a regulator to review, potentially providing greater confidence to the regulatory decision. Various participants expressed the need to integrate safety cases for operational and post-closure phases of a disposal facility. Situation-specific aspects for building a safety case were discussed and there was commonality in approaches, something further underscored in the poster session and other parts of this session.

The second part of Session III presented insights into the components of the safety case, including science and technology information challenges, knowledge management challenges and confidence issues. Methods; models; codes; databases; features, events and processes; and scenario management were all addressed. Discussion highlighted institutional components that help provide trust and confidence, one of which is an effective quality assurance programme. Another is a safety culture that empowers employees to question safety-related decisions and approaches; both of these points were endorsed during discussion.

The final part of Session III was diverse with an illustration of two independent probabilistic approaches used to provide confidence in modelling a disposal system, a regulatory view of the evolution of a safety case over time and a long-term perspective that can be adopted when nearly a century exists between the first and last safety case for a disposal facility, from before construction to final closure.

During various Session III sub-sessions, two presentations addressed developing additional performance criteria as a complement to the more traditional safety criteria of radiological dose or risk to a reference human being in the biosphere. Other country-specific papers provided regulatory perspectives on specific disposal facilities and the roles of independent safety evaluations by regulators. In addition, one paper reported on preliminary conclusions from an international project regarding the regulatory review of safety cases in several European countries, which recommended early and continuing dialogue between regulators and implementers to create common understanding of technical information and regulatory expectations.

Session IV (see Appendix D for papers presented in this session) addressed the potential role of the safety case in societal dialogue. Three papers presented the perspectives and experience of potential host communities for disposal facilities, in terms of understanding and building confidence in the safety case, as well as regarding trust in the institutions involved. The communities engaged in dialogue and information exchanges with implementing organisations and regulatory agencies. The communities reflect different cultural values according to the national context. In addition, they are accorded varying degrees of autonomy or self-determination in decision making for the siting process, depending on the national legislative and regulatory context for radioactive waste disposal. The acceptance of a disposal facility by the communities was viewed to be affected by the role accorded to it in the decision-making process; health and safety were primary concerns and economic benefits were also a consideration. All three communities were provided an opportunity for information exchange and financial resources to help support such a dialogue. These communities have successfully developed and engaged expertise in the safety case to support a constructive dialogue on issues related to safety. Optimism was expressed over the ability of laypersons to understand a safety case, if communicated clearly.

A discussion session ended Session IV, which addressed: (1) how the safety case becomes embedded into societal processes of establishing disposal programmes, (2) the perception of timescales and risks by laypersons, and (3) the division that may exist within the same agency if technical experts performing safety assessments do not communicate well with those who are working in stakeholder involvement. To make a safety case understandable to wider audiences, a programme needs to bring together both technical and social sciences. There are vast differences between perceptions of risks by the general public and by technical experts developing the safety case and strictly numerical assessment of risks does not distinguish between voluntary and imposed risks, an issue of importance to laypersons. Some aspects which may engender scepticism or mistrust in the public are, for example, statements about safety over extremely long time frames and the assumption, made for the purpose of performance assessment, that the disposal system must be shown to function without human intervention after a relatively short time (which may be mistakenly interpreted as allowing or encouraging abandonment of the site by those responsible). It is not clear how well regulatory requirements for the extremely far future serve societies; while scientists, and especially geologists, are accustomed to working with timescales of thousands to millions of years, claims to able to assess safety over such times may appear to the public to be implausible or arrogant.

The discussion addressed also the perceived difficulty for regulators and implementers to acknowledge that their decisions are also societal decisions and are, thus, essentially "co-decisions" with other decision-making entities in society. It is not always clear how scientists (and scientific analyses) fit into this social dynamic. Policy decisions prescribed by law and regulation reflect societal values or the power base that exists within the country. Nevertheless, people want to know that sound scientific studies have been performed even if they do not understand fully the technical information and they want assurance that competent authorities believe that there will be no release of radionuclides.

The poster session was presented between Sessions III and IV (see Appendix E for papers presented in the poster session). The posters displayed a variety of papers suggesting that a basic universal set of activities exists in the pursuit of designing and building a safety case. The common activities require an approach that includes site characterisation and data collection, scenario development and modelling and the search for analogue insights and other ancillary sources of potentially supportive information. Also discussed in the poster session were similarities and potential overlaps between pre-closure and post-closure safety cases, the need to evaluate different processes at different times, and insights into specific processes of interest in evaluating safety.

After an overview of the session topics and highlights from throughout the three days of the meeting, the symposium was closed with statements from the NEA as well as by the cooperating agencies, the EC and IAEA. The symposium was recognised as a continuation of, and an important contribution to, international co-operation and discussion that has been ongoing over many years regarding the safety case. The development of common concepts, principles, terminology and approaches for the safety case has advanced considerably and further harmonisation may be achieved as collaboration continues through tools such as the IAEA Joint Conventions, EC projects, and the ongoing work of the NEA Integration Group for the Safety Case.

At that, the symposium was closed.

DEVELOPMENT AND MATURITY OF THE SAFETY CASE: AN INTERNATIONAL PERSPECTIVE FROM THE NEA SECRETARIAT

This section describes the evolution and development of the concept of the safety case, provides some perspective on the maturity of the concept in practice, and highlights key outcomes and conclusions of the symposium.

Background

Disposal of long-lived radioactive waste in engineered facilities or repositories, located in suitable deep underground geological formations, is being investigated worldwide to protect humans and the environment both now and in the future. In the specific field of safety, as of the 1990s, the concept of the safety case was developed and presented in *Confidence in the Long-term Safety of Deep Geological Repositories: Its Development and Communication,* published by the OECD Nuclear Energy Agency (NEA).

There has been notable convergence in documents published nationally and internationally on the understanding and development of long-term safety cases for geological disposal. Some examples are:

- The 2006 International Atomic Energy Agency (IAEA)/OECD-NEA publication of a safety standard report for geological disposal: *Geological Disposal of Radioactive Waste, Safety Requirements,* IAEA Safety Standards Series No. WS-R-4, Vienna, Austria.
- The 2004 NEA brochure *Post-closure Safety Case for Geological Repositories: Nature and Purpose,* NEA-03679, which defines and identifies the purpose and general contents of safety cases for geologic repositories for long-lived radioactive waste.
- Presentations at the international conference “Geological Repositories: Political and Technical Progress” held in Stockholm in December 2003.
- NEA peer reviews of recently released national geologic disposal programme reports, which show a significant assimilation of the principles by national organisations.

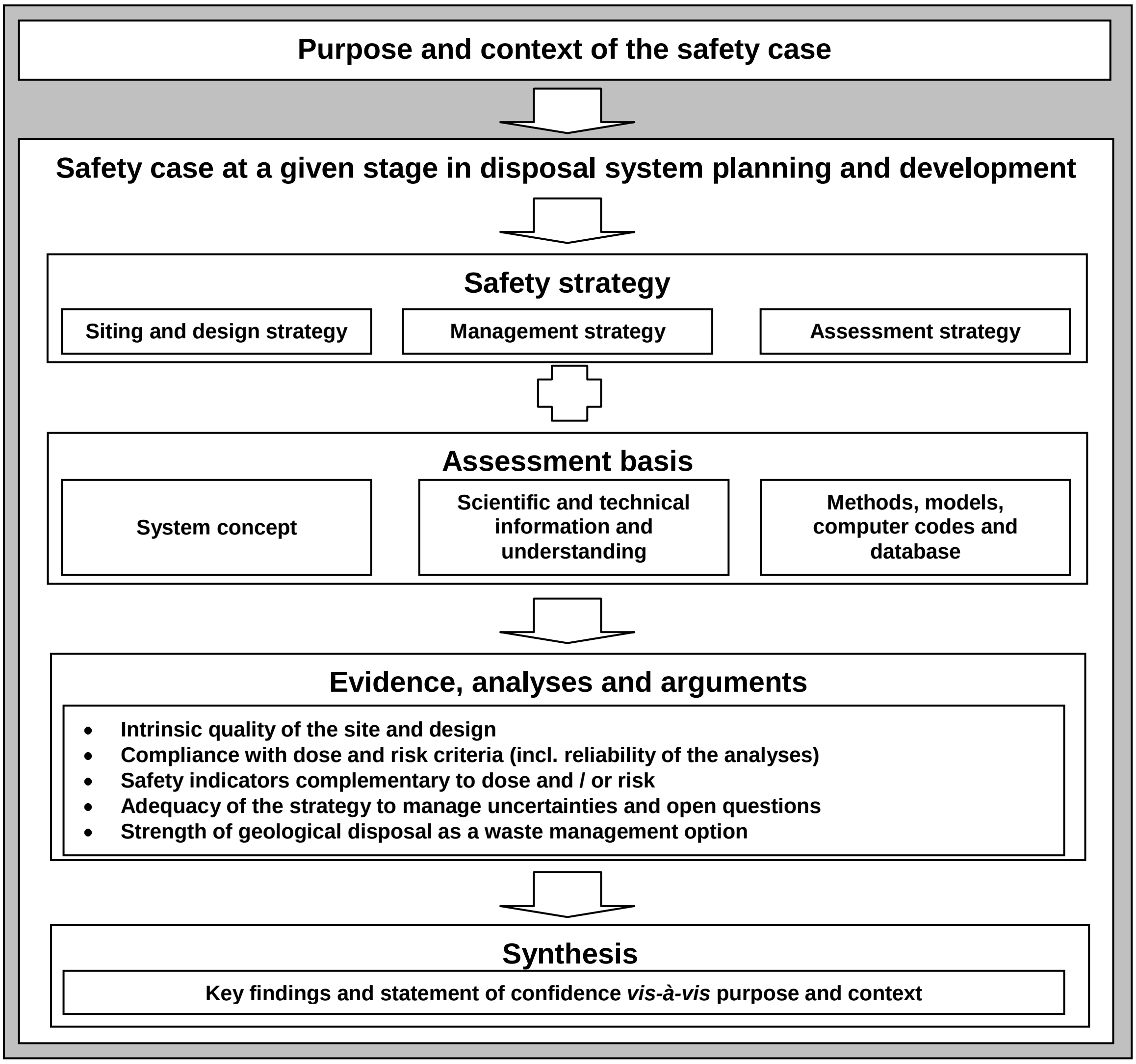
This symposium intended to provide a point of reference for those involved in the development of safety cases and for those with responsibility for, or interest in, decision making in radioactive waste management. The symposium sought to take stock of progress in member country programmes under the safety case concept and elements, championed within the NEA and addressed by international working groups within the IAEA and the European Commission (EC).

The concept and meaning of the “safety case”

A safety case for the post-closure phase of a geologic repository is a synthesis of evidence, analyses and arguments to quantify and substantiate that a repository will be safe after closure and beyond the time when active control of the facility can be relied upon. A safety case is presented, most often by organisations responsible for implementing waste disposal solutions, at specific points in the process of repository development. A safety case is typically used to support a decision to move to the next stage of repository development, but it could also be prepared to help review the current status of a project or in view of testing the methodology for developing a safety case. A key function of the safety case is to provide a platform for informed discussions whereby interested parties can express their level of confidence in a project at a given stage, including any issue upon which further work is warranted. This continued process of review and development is expected to result in a comprehensive and cogent safety case and in high, shared confidence in the quality of the decision it was meant to support. Safety cases are prepared throughout the lifetime of a repository and are often planned to continue even after closure, which requires an important and continued effort of updating, reviewing and revising on the part of regulators, implementers and other societal groups. This implies also a need to establish and utilise appropriate means to manage and transfer information over decades and even across generations.

The elements of the safety case, as described in *Post-closure Safety Case for Geological Repositories: Nature and Purpose* (OECD/NEA, 2004), are depicted in Figure 1.

Figure 1. **An overview of the relationship between the different elements of a safety case**



A clear statement of purpose and context is an intrinsic part of the safety case. In addition, recognising that format and content should be adapted to the decision context of each safety case, elements that contribute to the safety case are the safety strategy, the assessment basis, the evidence and arguments for safety and a synthesis statement.

The safety strategy is the high-level approach adopted for achieving safe disposal, including an overall management strategy, a siting and design strategy and an assessment strategy. The strategy aims to incorporate good management and engineering principles and practice, and to provide sufficient flexibility to cope with new information as well as to take advantage of advances in scientific understanding and engineering techniques. Siting strategies tend to favour robustness and minimise uncertainty through the selection of a geological setting with assessable features. The assessment strategy must ensure that safety assessments capture, describe and analyse factors and conditions that are relevant to safety, and investigate their effects.

The assessment basis is the collection of information and analysis tools supporting the safety assessment. This includes an overall description of the disposal system that consists of the chosen repository and its geological setting; the scientific and technical data and understanding relevant to the assessment of a system-safety; and the assessment methods, models, computer codes and databases for analysing system performance. The quality and reliability of a safety assessment depends on the quality and reliability of the assessment basis.

Evidence, analyses and arguments for safety must be compiled into a safety case. Results of analyses are typically compared against safety criteria, often in terms of radiological dose and/or risk, but there may also be other performance measures applied either for regulatory compliance or as indicators of performance that provide insights into system behaviour. The evaluation of these performance measures or indicators, using mathematical analyses, is typically accompanied by more qualitative arguments that provide a context or support for the performance-calculation results. A series or range of appropriate evolution scenarios may be addressed for the disposal system. Robustness of the safety case may be strengthened by the use of multiple lines of evidence, leading to complementary safety arguments, to compensate for any shortcomings in confidence in any single argument.

The synthesis of available evidence, arguments and analyses, supported by the quality and reliability of the assessment basis, support a safety case statement of confidence, typically made by the implementer, that sufficient confidence exists in the safety of the system to justify a positive decision to proceed to the next stage of planning or implementation of a disposal system.

Perspective on the term "safety case"

The English word "case" can be interpreted in a legal sense, as if a case for repository safety were being prepared to be brought to trial. In such a trial, all relevant information for determining the safety or lack of safety of the proposed system would be brought to the attention of judges or juries, or both. The purpose of a safety case depends on the stage of the repository programme under consideration and the intended audience. In staying with the legal sense of the word "case," judges could be anyone from internal organisational decision makers to a national governing body or regulator. The jury, depending on the nature of the decision that is being supported by the safety case, can range from internal managers to the public and to external organisations that are either opposed to or perhaps proponents of the proposed geologic disposal facility (i.e. interested external parties commonly referred to as stakeholders). The word case has been judged too legalistic by several non-English-language national implementers or regulators, and does not necessarily reflect the way that decisions are made in some programmes or countries. The French, for example, use the word "dossier" to capture the idea of a compilation of information relevant to judging safety. Others have called preliminary safety cases either “safety evaluations” or “safety reports”. Regardless of what label is attached to such reports, evaluations, dossiers or cases, their importance is whether their content reflects the principles defined by international organisations as being necessary to provide as complete an argument for safety, and as complete a safety case as is needed, for the intended audience and purpose of the document.

Perspective on the maturity of the safety case concept

The safety case concept is generally understood, accepted and adopted by radioactive waste management programmes worldwide. It is recognised that the safety case concept includes more than calculated numerical results (in terms of radiological dose indicators, for example) to demonstrate safety or regulatory compliance. Specifically, the safety case also involves presenting underlying evidence supporting calculated results and providing decision makers with a basis for judging whether sufficient confidence exists in the safety evaluation to support a decision. At the symposium, a number of national geologic repository programmes presented papers on safety cases showing that the concepts considered important to making a case for safety are being seriously addressed. In addition, some national programmes in earlier stages of development have presented papers focusing on a specific aspect or aspects of the scientific or modelling basis in anticipation of making part of a safety case. These presentations indicate a natural progression that programmes go through to reach maturity in terms of the safety case concept promoted by the NEA, the IAEA and the EC.

A safety case must address both the scientific basis and the design basis for a disposal system. It takes time and resources to develop a basis that is both science-based and engineering-based. Furthermore, modelling capability must be sufficient to integrate the scientific and design basis for the proposed repository. This capability requires additional time and resources to develop, test and refine the capability. Safety evaluations performed at various stages are used to refine experimental approaches, designs and modelling tools. Over time, these evaluations mature into safety cases. The breadth of maturity displayed in each presentation of this symposium showed a healthy state of affairs with different programmes in different states of maturity benefiting from the insights and experience of others, all working to approach the same goal of making a competent case for safety that can withstand internal as well as external critical evaluation.

Outcomes of the symposium

Numerous attendees of the 2007 symposium recognised that safety cases have evolved into tools to both build confidence in safety and aid in decision making. Key aspects of this evolution include:

- Improved and structured documentation to favour clarity and traceability of argumentation.
- Argumentation that showcases the knowledge basis accumulated by the project.
- Development of more sophisticated analytical tools and databases.
- The introduction of new conceptual tools such as safety functions, which embody key aspects of performance of the geological disposal system and from which can be developed internal requirements that relate the ability of the disposal system to fulfil these functions, thus making more transparent the role of various components (and their synergies) in the disposal concept.
- Utilisation of performance and safety indicators besides the traditional radiological dose and risk indicators.
- Open discussion – in the safety case – of extant issues of concern and the identification of a path forward to resolution.

Examples of successful uses and progress of safety cases for decision making were presented by a number of national programmes. Additional lessons learnt include:

- International organisations play an important role in the formulation of the present concept of a safety case and in the development of methodologies that support it, as well as in the provision of peer reviews, which are an important means to ensure and increase the quality of safety cases.
- Dedicated experts from several disciplines that have integrated into teams are paramount to planning and developing a safety case.
- A broad range of aspects of the safety case can be discussed and refined with the help of local stakeholders. Notably, if a repository-hosting locality is large, there are likely to be citizens who have knowledge to competently review and comment on technical aspects of the safety case. In other cases, if resources are made available, independent consultants can provide the necessary expertise on behalf of a hosting community.
- Updated safety cases may need to be prepared at time intervals up until (and sometimes even after) a repository is closed. Successive safety cases could span a period of several decades up to centuries. Constant care must be exercised to preserve key data and ancillary information that establish the quality of those data.
- There is a good shared understanding of what a safety case is. The term “safety case”, however, is difficult to translate from English into other languages. Similar translation difficulties are met with other terms, such as safety versus security and safeguards, or confidence versus trust.

The 2007 symposium focused on the specific subject of the safety case for disposal. The symposium served a community of specialists working to establish state-of-the-art programmes in the safety case area and interested stakeholders. This symposium also provided verification that a shared view on the purpose and contents of a safety case serves to advance the discussion and exchange of experience. Given the importance of shared information and the heightened importance of safety cases for decision making in national programmes, a higher frequency of symposia could be warranted.

EXTENDED SUMMARY OF THE SYMPOSIUM

For each symposium session, a synopsis of the presentations and highlights of any following discussion are presented below. The synopses are based on the contents of presentations and on rapporteur reports prepared on each session in co-operation with the session chairperson.

The symposium consisted of four main presentation and discussion sessions, one poster session, a fifth session to summarise the symposium, and closing remarks by the three sponsoring international agencies as follows:

- **Session I: French National Programme on Deep Disposal Safety**, provided a comprehensive overview of recent progress regarding the safety case concept in the national programme for the symposium host country, France.
- **Session II: The Safety Case Concept and its Evolution**, provided an overview of the international evolution of the safety case concept. It described the progression of international guidance on the topic and provided key examples of implementation of these concepts in various national programmes. A panel discussion, Session II-c, explored these issues in more depth.

  Session II, therefore, consisted of three sub-sessions:
  - Session II-a: The International Evolution.
  - Session II-b: The National Evolution.
  - Session II-c: Plenary Panel Discussion.
- **Session III: Recent Experiences in Developing a Safety Case**, highlighted recent practical experiences in developing, supporting, and reviewing safety cases, including views of regulatory agencies and decision-makers.

  Session III consisted of three sub-sessions:
  - Session III-a: Approaches to Achieving Safety.
  - Session III-b: The Scientific and Technical Basis for Assessing Safety.
  - Session III-c: Evaluating and Documenting Confidence in Safety.
- **Session IV: The Embedding of the Safety Case in Societal Dialogue and Decision Making**, addressed the role of the safety case in societal dialogue and focused on communication aspects with the emphasis on non-institutional groups.
- **The poster session** presented additional papers on relevant topics during the second day of the symposium.
- **Session V: Report on the Meeting and Plenary Discussion**, consisted of an overview discussion of themes, progress and conclusions reflected in the symposium and became the basis for the executive summary.

The full papers submitted by symposium presenters are included in five session-specific appendices (Appendices A through E) of these proceedings.

Welcome addresses

The symposium opened with welcoming addresses by:

Organisation for Economic Co-operation and Development/Nuclear Energy Agency
G.H. Marcus, Deputy Director-General

French Ministry of Research
D. Goutte, Director, Department of Energy, Transport, Environment, and Natural Resources

International Atomic Energy Agency
E. Amaral, Director of Radiation, Transport and Waste Safety

European Commission
S. Webster, Head of Unit, Nuclear Fission and Radiation Protection of the Directorate General for Research.

Session I: French national programme on deep disposal safety

Session I was chaired by H. Riotte and the rapporteur was A. Hooper. This session dealt with developments and future plans in the French national programme from the perspectives of policy, implementation and regulation. Summaries of the presentations are provided as follows:

The Program Act of 28 June 2006 on sustainable management of radioactive materials and wastes

C. Vincent (Head of Sub-Directorate for Nuclear Industry, Directorate for Energy and Mineral Resources, General Directorate for Energy and Raw Materials, Ministry of Industry, France)

This presentation focused on the key aspects of the Programme Act of 28 June 2006 with respect to taking forward radioactive waste management policy in France, namely: (1) implementation of comprehensive national policy on radioactive materials and waste management, (2) reinforcement of transparency and democratic decision-making requirements and (3) organisational and financing dispositions. The law mandates creation of a national plan, to be updated every three years, for management of all radioactive materials.

The principles of protection of human health and safety and the environment as the responsibility of today's generation have now been deemed to apply to all radioactive materials. The three axes of research – partitioning and transmutation, deep geological disposal, and storage – established in the "Bataille" legislation of 1991 remain, but greater emphasis is now given to geological disposal with the other two areas being developed to support this end point. The national plan should show how the hierarchy of waste management principles are to be implemented and in particular how an integrated strategy is being developed; for example, how interim storage complements eventual disposal.

For geological disposal, the policy is to move from the past research and development phase to proactive implementation of a reversible disposal concept with goals to achieve review of an application for authorisation (i.e. license application) by 2015 and to begin operations in 2025.

With regard to transparency and social debate, the prohibition on disposal of waste from other countries or resulting from treatment of spent fuel or waste from other countries in a French repository was emphasised, as was the greater independence of the National Evaluation Commission, the future roles of government and the Parliament in the disposal facility start-up authorisation process and the strengthening of the competencies of the local information and monitoring committee.

The establishment of a dedicated fund to finance radioactive waste management under the new law was described, as was the creation and disbursement of additional taxes to finance research on deep geological disposal, and storage and economic support for local communities and technological diffusion. The governmental and parliamentary oversight model to ensure financial provisioning by operators for the long term was outlined as the final element to ensure safe disposal in the future.

French radioactive waste disposal programme: main advances since 1991 – way forward to 2015

T. Labalette (National Agency for Radioactive Waste Management, Andra, France)

In this presentation, a summary was given of the organisational structure and responsibilities for high-level radioactive waste (HLW) management in France, in particular the remit of Andra within the overall structure. Andra conducts research and serves as the implementing agency and operator for radioactive waste disposal facilities in France. The steps to deliver the implementation plan under the new law of 2006 were outlined.

For long-lived intermediate waste and HLW, it was noted that the 2006 law foresees a license application for geological disposal by 2015. Based significantly on the conclusions of *Dossier 2005 Argile* (required by the 1991 Bataille law and considered an important input to recent legislation), the focus is now directed to identifying an appropriate site. Particular emphasis was given to siting and to the transposition zone concept applied in the Meuse-Haute Marne region, which has led to the identification of an area of 200 km^2 where the properties of the Callo-Oxfordian argillites are equivalent to those at the Bure URL site.

A comprehensive geological survey programme will be undertaken in the near future to identify a smaller zone of about 30 km^2 for more detailed surveying using three-dimensional seismic surveys. Within this area, the potential sites will be identified and information on the site (or sites) will be provided for dialogue planned with the local community. The development of the future scientific and technical programme was also described, emphasising how the programme has been influenced by evaluations of *Dossier 2005 Argile*, which identified the need for work on topics such as the excavation disturbed zone, gas generation and migration, thermo-hydro-mechanical-chemical transients and coupling of phenomena. In moving the repository design to an industrial perspective, the Dossier 2005 design will be challenged and the design optimised. Information provided to the public will emphasise the role of demonstration, particularly of operational machinery.

French safety rules and review approaches regarding the geological disposal

P. Bodenez (French Nuclear Safety Authority, ASN, France)

This presentation introduced safety rules and review approaches, in particular the remit of the French Nuclear Safety Authority (ASN) regulator to:

- Establish the regulation for safety of long-term waste disposal.
- Authorise the different phases of the construction of the URL at Bure.
- Survey, from a safety point of view, the research conducted by Andra.

- Provide an opinion to the government on the steps established by the law, on the basis of reports from the Radiological Protection and Nuclear Safety Institute (IRSN) and opinions of a standing group of experts.

The current safety rule relevant to geological disposal (RFSIII.2.f) is under review but has been found to be broadly sound; updates are required to take account of the reversibility concept, of experience and feedback from France and abroad, and of new recommendations of the International Commission on Radiological Protection (ICRP), the IAEA and the NEA. The key radiological protection criteria are that, in the reference situation, the limit of individual dose is 0.25 mSv/a for a timeframe of 10 000 years, and after 10 000 years an evaluation of dose will be made, and 0.25 mSv/a will be retained as a reference value. The safety rule also sets objectives on the disposal concept, waste packages, engineered barriers and geological barriers with essential criteria on the stability of the site and hydrogeology. The move in philosophy to consideration of safety functions, rather than multiple barriers per se, was emphasised. The safety evaluation conducted by the regulator considers the following:

- Justification of any advantageous feature on the performance of each barrier.
- Evaluation of effects of disposal on the host rock and verification of the acceptability of these effects.
- Evaluation of the future behaviour of the disposal system and verification that individual doses meet radiation protection criteria.

The role of the ASN in conducting inspections at the Bure URL and in promulgating its opinion of 1 February 2006 on geological disposal was outlined. Of particular importance was the conclusion that it was reasonable to search for a site in the transposition zone around the Bure URL and that safe disposal is likely to be demonstrated in this area. This is complemented by the identification of additional studies that the ASN believes are necessary.

Recently, the ASN has authorised Andra to pursue research in the Bure URL until December 2011. The future responsibilities for authorisations were outlined, including the role of Parliament in establishing the reversibility conditions of the repository. Reversibility, used herein, means that the emplacement of waste in the repository should be reversible over a timeframe of 100 years under the law. Parliament will have to set the final reversibility conditions in law prior to authorisation of construction. Accordingly, the draft law is expected in 2014.

The conclusions of this presentation summarised the whole session in that the 2006 law has established a well-defined process, and complementary organisational structures and responsibilities, to support the development of safe geological disposal in France. Given the importance of international co-operation, Dossier 2005 Argile has been translated into English to make it accessible to a wider readership. It is available on the Andra website at www.andra.fr.

Session II: The safety case concept and its evolution

Session II was chaired by C. Ruíz López (Nuclear Safety Council, CSN, Spain) and the rapporteur was P. De Preter (Belgian Agency for Radioactive Waste and Enriched Fissile Materials, ONDRAF/NIRAS). Speakers in Session II illustrated the development of the safety case concept by international organisations, and the practical implementation of the safety case concept by national implementers and regulators. The wrap-up sub-session sought to draw upon the experience of participants to explore further the concept of the safety case and to discuss issues such as the importance of certain elements and terminology, the need for harmonisation and areas for further co-operation by international, national and regulatory organisations with regard to the development, implementation, uses and reviews of safety cases.

Session II-a: The international evolution

NEA: The safety case concept, its history and purposes

C. Pescatore (Nuclear Energy Agency, NEA, France)

In this presentation, discussion focused on the evolution of the safety case concept. To begin, the following timeline was reviewed:

- The 1989 NEA symposium addressed readiness to evaluate repository safety: *Safety Assessment of Radioactive Waste Repositories*, Proceedings of a CEC/IAEA/NEA Symposium, Paris, France, 9-13 October 1989, OECD/NEA, Paris, France, 1990. (Commission of the European Communities, CEC, shortened to the European Commission, EC, in the remainder of this proceedings document).
- The 1994 GEOVAL conference emphasised that confidence building is possible, not validation in the classical sense: *GEOVAL '94: Validation through model testing*, Proceedings of an NEA/SKI Symposium, Paris, France, 11-14 October 1994, OECD/NEA, 1995.
- From 1995 to 1997, NEA sponsored the Integrated Performance Assessment Group (IPAG) and its three exercises.
- In 1998, Publication 81 of the ICRP, *Radiation Protection Recommendations as Applied to the Disposal of Long-Lived Radioactive Waste*, made recommendations on how to apply ICRP information to potential doses.
- In 2003, the EC published the **S**afety and **P**erformance **In**dicators (SPIN) report on performance indicators: *Testing of Safety and Performance Indicators*, EC Project FIKW-CT-2000-00081, Report EUR 19965, 2003, European Commission, Brussels, Belgium.
- In 2000, NEA created its Integration Group for the Safety Case (IGSC), which continues its work up through the present.
- In 2007, IGSC published its "Timeframes" report addressing issues of calculation and confidence in assessment over the various timescales of concern.

The basic concept of a safety case was not conceived in the radioactive waste disposal field, rather the concept was already used in the 1970s and 1980s as a tool in fields other than radioactive waste management. The application of the concept to radioactive waste disposal poses particular challenges and considerations are still evolving and being carefully evaluated (see also the plenary discussion in Session II-c regarding whether unresolved issues should be addressed in a safety case).

In international discussions, the NEA IPAG exercise introduced the term "safety case" (see terminology and definitions in Establishing and Communicating Confidence in the Safety of Deep Geological Disposal, Approaches and Arguments, Integrated Performance Assessment Group, IPAG, Report III, OECD/NEA, Paris, France, 2002).

The evolution of the safety case is a gradual and logical path from (1) performance assessment to (2) discussions at the international level of verification, validation or confidence building as applied to long-term safety to (3) development of the concept of a safety case at the national and international level and to (4) effective and decision-oriented use of this concept in national programmes. As national programmes advance towards implementation with the first steps of licence applications for construction and operation, the safety cases developed and submitted will also evolve in terms of systematic approach, level of detail, comprehensiveness and other activities. This evolution is related to aspects such as:

- Increased awareness of the fundamental difference between the quality of the system that has to provide safety and the quality of the information used to assess the safety of the system; linked to this is the distinction between the strategy to achieve safety and the strategy to argue safety.
- Growing importance of qualitative arguments (quality of site and design) alongside quantitative assessments and results as a way to evaluate and confirm quality; multiple lines of evidence, complementary arguments, and various safety and performance indicators play a role in these qualitative aspects of a safety case.
- Increased orientation of the work to decision makers and the needs of different stakeholders involved in the decision process.
- Increased emphasis on the managerial approach to ensure integration of information and arguments.
- Growing awareness of the importance of a systematic, transparent and traceable treatment of uncertainties.
- Confidence statements in a safety case – does a safety case really require an explicit statement of confidence? How to express this confidence is still a field of diverging points of view.

This evolution has been captured at the international level in IAEA and NEA documents, which consolidate the changing view of the safety case. This evolution constitutes the basis for national programmes to develop safety cases as decision-aiding and decision-making tools in a stepwise manner. A major achievement in this evolution may be the movement away from considering a performance assessment in the safety case to be sufficient, from a scientific and mathematical viewpoint, without addressing concerns from stakeholders and the needs of decision makers.

ICRP: The evolution of thought from the ICRP 46-concept of potential exposure

A. Sugier (Radiological Protection and Nuclear Safety Institute, IRSN, on behalf of the International Commission on Radiological Protection, ICRP)

This presentation began with an explanation of how ICRP Publication 46 (ICRP 1985), which is now considered too theoretical, had led to an understanding that the assessment and judgment of long-term safety cannot rely solely on quantitative approaches. The two premises underlying current work by the ICRP are:

1. There is a continuum of risk.
2. The risk is dependent on the context of exposure.

This second premise is essential for the discussion and regulation of long-term safety and emphasises the importance of the societal dimension when dealing with protection of future generations living thousands of years in the future. In the new draft recommendations made by the ICRP, three types of exposure situations are defined:

- Planned exposure situations: planned introduction and operation of sources or planned work with sources including potential exposures.
- Emergency exposure situations: unexpected situations, requiring urgent action, that occur during the operation of a practice.

- Existing exposure situations: situations that already exist when a decision on control has to be taken, including natural background radiation and residues from past practices.

Disposal facilities are seen as planned situations. For all types of situations, the same basic approach of radiation protection applies, which is an important difference from the current recommendations in which “practices” and “interventions” are dealt with differently. The revised ICRP approach acknowledges that, in reality, “practices” and “interventions” are intertwined in a process of optimisation that aims at exposures below the reference levels or dose constraints, as much as is reasonably achievable.

For long-term safety of disposal, dose can only be used as an indicator, and dose calculations cannot be the sole element of the decisions. In the assessment of radiological impacts or exposures, the fundamental distinction is between natural processes (both gradual and disruptive), which can be assessed on the basis of available knowledge of the underlying processes and events, and human intrusion scenarios, which cannot be assessed in the same manner. Consequently, the concept of dose constraint does not apply to human intrusion. Additionally, an important concept for planned situations is that of potential exposures caused by malevolent events or deviations from planned procedures. In assessing potential doses, both the probability of the event and the resulting dose need to be considered. This concept is seen by ICRP as being applicable even for the far future. Guidance coming from the ICRP in its revised approach to radiation protection is expected to include the values and considerations listed in Table 2.

Table 1. **ICRP New Publication Projected Dose**

Bands of Projected Dose	Characteristics and Requirements
20 to 100 mSv	*Exceptional* situations. Benefit on a case-by-case basis. Information, training and individual monitoring of workers. Assessment of public doses.
1 to 20 mSv	*Individual* direct or indirect benefit. Information, training and either individual monitoring or assessment.
0.01 to 1 mSv	*Societal* benefit (not individual). No information, training or individual monitoring. Assessment of doses for compliance.

For some human intrusion scenarios, it is possible that the assessed impacts could exceed the levels at which ICRP recommends that some “intervention” may be justified. As a first step step, the levels of conservatism in the assessment should be checked. The second step in an intervention would be to consider the design of the system in order to reduce the likelihood of intrusion or to limit its consequences. The three characteristics of exceptional, individual, and societal potential exposures (Table 2) are planned potential exposures with considerable uncertainty regarding whether they would actually take place.

The same approach to “intervention” applies for existing disposal facilities (i.e. evaluate the situation in terms of radiological impact and assess what level and type of optimisation is required to remain under the reference levels). This is clearly an assessment where stakeholders have to be involved and where consideration has to be given to relevant socio-economical factors. Generally, intervention is indicated if dose levels are at 1 to 20 mSv/a. Intervention, in this context, may mean the redesign of a repository if socio-economic factors dictate against a repository that allows a plausible likelihood of dose levels in this range.

The discussion clarified some of the points made by the presenter. One controversial item was the categorisation by the ICRP of geologic disposal as a "planned exposure situation". It was pointed out that these terms could cause confusion to non-radiation protection specialists. Namely, while geological disposal is itself a "planned exposure situation", the exposures from geological disposal are "potential exposures" (exposures that may not ever exist over the regulatory timeframe). The issue of terminology and the substance of the problem would need more in-depth analysis by the ICRP and other interested parties. Ms. Sugier thanked the audience for the feedback and indicated that she would relay the messages given during the discussion to the ICRP main commission.

IAEA: The evolution of safety standards and requirements (WS-R-4) in respect of the safety case

P. Metcalf (International Atomic Energy Agency, IAEA, Austria)

This presentation described the evolution of IAEA safety standards and requirements pertaining to safety cases. Specifically, it was explained how the concepts of safety case and of safety assessment are addressed in WS-R-4, dealing with geological disposal requirements, co-sponsored by the NEA. A safety case is considered a central tool for stepwise development, operation and closure of a disposal facility. A safety case is decision-oriented, both at the technical level (design, regulatory compliance) and at the political level (a national decision for disposal).

During the symposium, the content of the WS-R-4 document, as follows, was discussed in some detail:

- Introduction.
- Safety objectives and criteria for operational and post-closure periods.
- Safety requirements.
- Appendix – assurance of compliance with safety objectives and criteria.
- Annex – geological disposal and the principles of radioactive waste management.

The safety requirements address the framework for geological disposal, the steps in the development of a facility and the safety assessment and safety case. The safety case is essential to decision making. The characteristics of a safety case substantiate safety and contribute to confidence. A safety case includes clear documentation of:

- Design and design logic.
- Supporting evidence and reasoning on the robustness and reliability of the facility.
- Output of the safety assessment.
- Quality of the safety assessment and underlying assumptions.
- Perspectives on the results of the safety assessment and issues to be resolved.

Justification, traceability and clarity are key features.

The WS-R-4 document formulates the requirements at a general level. The IAEA intends to develop more detailed guidance and recommendations in the underlying safety guidance documents (e.g. with respect to addressing the needs of different stakeholders).

EC: Lessons learnt from 25 years of experience with performance assessment/ safety assessment exercises in the European Union

J. Mönig (Gesellschaft für Anlagen- und Reaktorsicherheit mbH, GRS-Braunschweig, Germany), J. Marivoet (Nuclear Research Centre, Belgium, SCK•CEN), and J. Alonso (Empresa Nacional de Residuos Radiactivos, S.A., ENRESA, Spain]

This presentation provided an overview of the EC projects in the field of performance assessment and safety assessment. From 1982 to 1989, work began with the performance assessment of geological isolation systems, covering more than 20 years of research and development. Priorities, approaches, methods and themes and the issues dealt with in the EC projects are clearly linked to progress made in national programmes and to the evolution of safety-case concepts presented in the foregoing paper by the NEA. Early projects examined, for example, the feasibility of safe disposal in different types of geological formations and the variability introduced by biosphere modelling, whereas later projects evaluated additional safety and performance indicators and explored more detailed process modelling.

The importance of a common methodology and terminology that enables comparing and understanding the results obtained for different disposal systems was emphasised. Common, however, does not mean identical. The projects also highlighted that simpler performance assessment models can facilitate uncertainty analysis and improve transparency and communication, but the complexity must always be consistent with the available data basis. In addition, the EC projects emphasised that justification of basic assumptions is essential and can be improved by detailed process analysis.

A major added value of EC projects and of international collaboration is bringing together a critical mass of experts to make progress in the field of long-term assessments. The international safety assessment knowledge available can be used in national programmes through international reviews. Many national programmes already use the tools, concepts, approaches and methods developed and discussed at the international level.

Session II-b: The national evolution

A comparison of national dose and risk criteria for deep disposal of long-lived radioactive waste

R. Ferch (Consultant, Canada)

This presentation provided a comparison of national dose and risk criteria, beginning with the history of work performed by the NEA long-term safety criteria group to examine national criteria (radiological dose and risk) for long-term safety of geological disposal. This work has shown that there are wide variations in the numerical criteria (i.e. more than a factor of 20). Such differences, however, should be looked at in the broader frame of assessment approaches (e.g. conservative, bounding and realistic) and compliance judgments (e.g. hard limits or targets, timeframes, and other considerations). A direct comparison of numerical criteria and results of safety assessments can be misleading without consideration of the scope and construction of the underlying assessment. Whether a need exists for harmonisation of these criteria was discussed during the panel discussion (Session II-c).

Additionally, regulations not only focus on dose or risk, but also address other complementary indicators such as radionuclide fluxes and other concepts and design-based criteria like “best available technology” to judge the safety of a disposal system.

The question is: To what extent do different criteria reflect different fundamental objectives of long-term protection where ethical and societal factors are considered? This issue is seldom discussed, or so it appears, at the highest political level in countries (e.g. at the parliamentary level) and it is not always clear to what extent the criteria reflect societal or political consensus on the fundamental objectives of protection in the far future. International exchanges might be useful here to avoid regulations that might reflect the opinion of a only few persons within a specialised agency.

The presentation included a listing of dose/risk constraints by country. Some criteria were expressed as dose constraints (i.e. targets rather than hard limits). Also, the assessment approach, whether it sought realism or a bounding estimate, may mean that the observed differences in standards are limited. It was reiterated that caution should be used in comparing criteria and corresponding performance assessment results across national boundaries, given the diversity of regulatory approaches considering: (1) limits versus constraints, (2) realistic approaches versus bounding approaches, (3) the scope of events and processes included or excluded and (4) other factors. Considerations other than dose or risk projections are involved in making comparisons.

NAGRA – How has the safety case concept evolved in the Swiss national programme?

P. Zuidema and J. Schneider (National Cooperative for the Disposal of Radioactive Waste, NAGRA, Switzerland)

This presentation described the evolution of the safety case concept and its implementation in the Swiss national programme. The presentation made an important distinction between the meaning of "quality of the system" and "quality of the understanding". That is, even when the disposal system concept, site, and design remain constant (i.e. the quality of the system, or its inherent safety), the quality of the understanding can be improved by research and development activities, which should be reflected in safety cases. The experience of the Swiss programme demonstrates improvements over time in the quality of the understanding. The presentation, however, also showed that changes to the system (e.g. choice of alternative host rock or modifications of the engineered barrier system) due to feedback from the safety case can be an important element in the development of a repository. The evolution in the Swiss HLW programme, which has produced safety cases in 1985, 1994 and 2002, also reflects evolution of the safety case concept and the scope of its elements on the international level, as described in the NEA presentation of Session II-a.

The safety case programmes of the Swiss, French, Swedish and Finnish, clearly show that the safety case is of high importance for key decisions in national programmes. These key decisions were only possible because sufficiently "mature" safety cases were developed and presented at key decision points, and these safety cases correctly addressed the needs at that particular point in the programme.

A safety case is a combination of both quantitative and qualitative arguments. The elements of a safety case span a broad range of activities, from model verification (main emphasis in the early phases of safety analysis) to confidence building (key area in recent developments in the safety case), and reflect the corresponding evolution of international activities. Emphasis should not only be on consequences but also on an analysis showing where the majority of radionuclides stay in the system.

The experience of the Swiss programme highlights that key features of a successful safety case include:

- The importance of the scientific basis (e.g. what is known? what is not known?).
- Systematic processing of information (e.g. complete, unbiased and balanced) with an adequate tool box (e.g. methods, codes and computers).

- Assessment of uncertainties.
- Feedback (within the current phase; also as input for the next phase).
- Summary (quantitative and qualitative) of key findings (quality of the system and quality of the understanding).
- Well-structured documentation (i.e. transparency and traceability).
- An integrated and dedicated team.

A case study in evolving regulations: proposed amendments to the environmental radiation protection standards for Yucca Mountain, USA

B. Gitlin (United States Environmental Protection Agency, U.S. EPA, United States of America).

In this presentation, the EPA explained the current status on its proposed amendments to U.S. radiation protection standards for disposal at the Yucca Mountain site. The EPA initially established such safety standards in 2001, including dose limits to (representative hypothetical future) individuals as well as ground water radiation concentration limits applicable for 10 000 years after closure. In 2005, the EPA proposed additional criteria for evaluating the results of safety assessments out to one million years after closure. The proposed 3.5 mSv/a dose standard for the Yucca Mountain repository beyond 10 000 years takes account of a recent court ruling and legislative dictates, as well as recommendations by scientific experts and international guidance. The court ruling rejected the previous EPA approach to continue safety assessment projections until peak dose, but to not apply a dose limit to the results of calculations (i.e. the use of complementary safety indicators or qualitative targets rather than a hard dose constraint is not an option). The proposed higher dose limit beyond 10 000 years recognises the inherent limitations in projecting performance in the very far future. It includes consideration of disruptive events (probability weighted).

The development of a final regulation based on the EPA proposal entails a process that includes public input through hearings and written comments. Thousands of public comments from elected officials to the general public nationwide have addressed numerous issues – including several outside the remit of EPA-ranging from general health concerns to highly technical issues. These issues include: rationale for the higher dose limit, including the use of background radiation as a basis; interpretation of ethical principles such as intergenerational equity; statistical measures used to judge compliance; and consideration of disruptive events. The EPA considers these comments in developing a final regulation and hopes to complete the new long-term standard within the next few months.

In discussion following the presentation, the speaker emphasised that the legislative mandate for EPA regulations at Yucca Mountain specifies the use of a dose limit as a safety indicator. As previously mentioned, a recent court ruling has effectively ruled out the possibility of using more qualitative indicators, even for extremely long time periods.

Session II-c: Plenary panel session

The moderators of this plenary panel session were M. Sailer (Oeko Institut, Germany) and C. McCombie (Arius and McCombie Consulting, Switzerland), and the rapporteur was R. Ferch (Consultant, Canada).

The moderators started the session by posing several questions for discussion, as follows:

- Do the presentations of national practices in Session II show important discrepancies?

- There are several different definitions of the term "safety case". Can there be agreement on one?
- Do national regulations have a strong influence on the safety case (e.g. variations in dose limits and treatment of time scales)?
- Is there a definition/quantification of the concept of "confidence"?
- How are open technical questions and uncertainties treated?
- Is a common international approach necessary, useful or merely confusing?
- How are multiple lines of evidence, analyses and arguments used?
- How should a safety case be presented to different audiences?
- What are the key similarities/differences between safety cases in geological disposal and in other areas or industries (e.g. oil and transport industries)?

The following discussion topics were also suggested by the audience:

- Harmonisation of safety criteria:
  - Does a lack of consistency among national criteria result in an untenable situation? Is harmonisation of terms and approaches necessary?
  - Is there a difference between harmonisation and consistency? If so, which is preferred?
- What is the relationship between experiments in underground research laboratories (URLs) and licensing safety cases?
- Is everyone comfortable with the use of ICRP concepts of planned versus unplanned exposure for calculated future doses for geological disposal?

The discussion first addressed the question of whether a safety case can have open issues when it is being used to "demonstrate" safety. The NEA definition of a safety case explicitly allows this and states that one role of a safety case, at earlier stages, is to identify issues that may be of concern or upon which further work is warranted. Nonetheless, some participants posited the view that if the purpose of a safety case is to demonstrate convincingly the safety of the repository, then the case could be undermined by the identification of unresolved issues, except perhaps those that demonstrably have no significant impact on safety. This position was refuted by several participants who argued that the purpose of the safety case must be viewed more broadly to include demonstrating how safety will be supported, including plans for the resolution of currently open issues. The majority of the audience appeared to support the broader interpretation. In view of the broader interpretation, the safety case is considered a "living" document that changes as the design evolves (e.g. demonstrating that the concept is a good one in early stages, progressing to demonstration that a safe system has been implemented).

It is expected in a stepwise decision-making process that not all issues can be identified and resolved in the earliest stages. Information gathered during the initial stages is expected to contribute to later decision making and, therefore, may have an impact on the safety case as it evolves. It was also proposed that during the early stages, a safety case that raises and discusses open issues indicates a successful safety culture on behalf of the implementer, whereas a safety case that appears to claim that all issues have been resolved may actually raise questions of trust in the implementer. Those who expressed that a safety case may not have open issues acknowledged that the case could still put forth assumptions that call for further evidence to be provided in later stages, to fully support confidence in long-term safety. Thus, whatever terminology is applied, as more information becomes available over time, the safety case is refined to support the next decision point.

The approach of a living safety case poses the challenge of explaining to a non-technical audience how a safety case that is admittedly and explicitly incomplete can still provide sufficient support for decisions of approval in the early stages of a project. While a scientific audience might welcome a safety case that raises issues, and explains the measures to be taken to resolve them, the public and politicians may want an early-stage safety case that demonstrates safety to a high level of confidence before making the investment needed to allow a project to proceed.

The discussion then turned to the question of whether it is necessary or desirable to aim for harmonisation and consistency among differing national standards and regulatory criteria. There was a fairly broad range of opinions. Inconsistency among standards can lead to a perception that the level of protection provided by one programme is weaker than others or that safety issues have not been resolved. The cautions that numerical standards cannot be compared directly are not well appreciated outside the technical community. Interlocutors point out places where the standards applied to a proposal are "weaker" than those applied in some other country and use that as a criticism, in some cases, to the point of claiming that the criteria were adjusted to suit a particular project proposal.

Some technical difficulties stand in the way of harmonisation. Different geological settings often result in differences in the relative importance of various scenarios and sources of uncertainty, which a "one size fits all" approach would not accommodate. Even systems that might be similar in terms of operational considerations could involve disposal concepts and geological settings that evolve very differently over time; thus, notwithstanding that comparisons might reasonably be made in the near term, they may become inappropriate and insupportable as the time scale increases.

To achieve consistency, it is not enough to harmonise dose and risk criteria. Other criteria such as performance expectations for safety functions, as well as ways with which uncertainties and conservatism are dealt, would have to be harmonised. It might be more important and fruitful to reach agreement on methodological issues and on fundamental science, rather than establishing consistency in dose or risk limits. Discussion on the difficulty of harmonisation included concerns that were social or cultural in nature. For example, existing criteria and approaches might be the result of a particular historical development in a country, for which a common standard could be inappropriate. There is evidence of national differences in public expectations about such items as retrievability and acceptability based on cut-offs in time or on low probability.

In general, reaching consistency among criteria and approaches would be a very difficult task to achieve, possibly requiring agreement at the level of a new international convention. Thus, investing efforts on a more achievable goal might be more reasonable. Perhaps, for example, there is or could be fundamental consistency at the level of underlying safety goals, even if the legal and political frameworks in which these were expressed resulted in differences at the level of regulatory criteria. It was pointed out, however, that agreement at this level, or focusing on scientific and methodological issues, would run the risk of being understood only by a technical audience.

The session concluded without reaching a clear consensus on the issues of harmonisation and consistency. Time constraints of the symposium did not allow for other suggested issues to be addressed.

Session III: Recent experiences in developing a safety case

Speakers in Session III provided illustrations of practical experiences in developing safety cases as well as implementation of research and development when building a safety case and when reviewing a safety case (e.g. regulatory perspectives). The sub sessions sought to illustrate three distinct aspects of the programme to prepare a safety case: (1) setting of stage for a programme seeking to build a safety case, (2) the types of technical information needed for a safety case and (3) approaches to building confidence to support a programmatic or societal decision in light of inevitable uncertainties.

Session III-a: Approaches to achieving safety

Session III-a was chaired by P. Zuidema (NAGRA) and the rapporteur was S. Voinis (Andra). Presentations in this subsession illustrated practical experience in developing safety cases in terms of siting and design strategies, management strategies and assessment strategies.

Introduction: Review of the preliminary results of the NEA survey on international experience in the safety case (INTESC)

J. Andersson (JA Streamflow AB)

This presentation on the NEA project on International Experience in developing Safety Cases (INTESC) provided a detailed overview of how the safety case concept has been applied or will be applied in NEA member countries. The safety case concept can apply to both generic and site-specific cases. The level of detail depends on the stage of a safety case in a national programme. This presentation gave preliminary findings based on responses, by 16 organisations from ten countries, to a detailed questionnaire exploring how programmes interpret and implement major elements of the safety case.

General agreement has been achieved on the safety case concept and elements as outlined in the NEA post-closure safety case brochure (NEA, 2004). Organisations that are more advanced in practical applications have produced individual safety cases and safety-case descriptions that have contributed to international discussion. Contributions have been made by France, the United Kingdom, Belgium, Switzerland, and others regarding post-closure safety, and some have also extended the safety case to address pre-closure safety for a facility, depending on the stage of the programme, as well as other factors such as whether regulations treat operational safety alongside or separate from long-term safety.

Various practical approaches are demonstrated, especially with respect to the relevance of defining safety functions, and the availability of qualitative and quantitative analyses. The role of the NEA features, events, and processes (FEP) database is transitioning to a completeness-checking loop, in particular for more advanced organisations; whereas in early approaches to safety cases the database is often a starting point for examining and defining FEPs.

It was noted that the more a repository programme progresses, the more the safety case is detailed and the more the design may be optimised. This is clearly an iterative process; while feedback from safety cases is used to refine the design, it is conversely also true that a more complete design allows a more complete, the safety case becomes more detailed as further information is available and assumptions can be refined.

Bátaapáti repository (Hungary): Evolution of the safety case

M. Goldsworthy, P. Molnar, and G. Danko (Golder Associates); Z. Nagy and F. Frigyesi (Public Agency for Radioactive Waste Management, PURAM, Hungary)

This presentation discussed the evolution of the safety case for the Bátaapáti Repository. This repository is proposed for low-level and intermediate-level radioactive waste. A crystalline upland site, at 200 to 300 m elevation above mean sea level, has been identified for characterisation. The site is a loess-covered topography and access roads are not yet fully completed. There is limited borehole data. There has been a history of public acceptance issues. From 1980 to 1995, the Hungarian repository development programme followed a stepwise process aimed at choosing the repository concept (i.e. deep or shallow disposal), site screening and gaining public acceptance. During this period, a first site was rejected, but it is not clear that the decision was explicitly a rejection of the proposal, but rather could reflect broader issues and sentiment regarding government control. From 1996 to 2000, the situation evolved considerably, with the decision to proceed with characterisation of the Bátaapáti site for a potential deep geological repository and the establishment of a stable institutional framework based on a new "Atomic Law" and founding of the Public Agency for Radioactive Waste Management (PURAM). Design and performance assessments were conducted as site investigations progressed during this timeframe.

Since 2001, the geological suitability of the site was formally confirmed and site investigations have continued both at the surface and underground. By 2004, three boreholes showed evidence of inhomogeneity in flow properties, "old" groundwater and downward flow. In 2005, a conceptual design was developed, with international input, for disposal in monzogranite with cement backfill. Continuing investigations on these bases have led to a more sophisticated understanding of complex high permeability and barrier zones. Discussion suggested that this is not unusual. Gathering more data may show greater complexity at depth in a natural system than is evident based solely on surface inspections and studies.

An iterative approach to achieving safety: Application in the "Dossier Argile 2005"

B. Cahen, S. Voinis, and L. Griffault (Andra, France)

This presentation illustrated that the safety case is an iterative process within a stepwise repository programme (see T. Labalette *et al.* presentation of Session II) and demonstrated how the elements are developed and connected. The "Dossier Argile 2005" reflected the feasibility stage for Andra's programme. Technical options using available technologies were analysed based on real site data, but the disposal system design is not optimised. The dossier considered pre-closure and post-closure safety; the assessment for pre-closure safety was based on a more classical "safety analysis", consistent with nuclear industrial facilities practices. The repository conceptual design relies on safety functions that were identified in relation to time periods. In this iterative process, managing uncertainty is a key part of the safety case (qualitative analysis) and there also is a key connection with safety analyses providing feedback to help direct engineering and research studies. The preliminary safety case in "Dossier Argile 2005" provides a reasonable degree of confidence and identifies the needs to be addressed in a more detailed safety assessment; the safety case makes both qualitative and quantitative arguments for the project to move ahead.

There was discussion of the differences in levels of information and detail between the "Dossier Argile 2005" and the dossier devoted to crystalline rock, which is considerably less detailed. The crystalline rock was considered but work stopped a few years ago because of both political opposition and technical objections to the challenges posed by overlying sediments. The "Dossier Argile 2005" is, therefore, more detailed and based on site characterisation. In response to questions, it was clarified that the area and exact shape of the footprint of the repository would depend on the waste streams eventually designated to be emplaced. Spent nuclear fuel or HLW would have different size requirements, depending on their activity level, requiring several square kilometres at a minimum. Certain wastes would be placed in different zones to manage thermal-hydrogeologic-mechanical-chemical interactions.

Developing a safety case for Ontario Power Generation's L&ILW deep geologic repository (DGR)

H. Leung, P. Gierszewski, T. Kempe, R. Heystee, and M. Jensen (Ontario Power Generation, Canada)

This presentation made a clear point that safety cases are iterative and that early iterations will identify open issues needing more work. The safety case for the deep geologic repository (DGR) in Canada is consistent with the NEA safety case approach. A facility was described, located in limestone host rock of the Michigan Basin, proposed for low-level and intermediate-level radioactive wastes only. From site characterisation, the host rock is considered stable. A question was asked whether the high salinity of groundwater encountered presented an issue with respect to performance; for the DGR, this is actually viewed as an asset because it demonstrates the stability of the groundwater in terms of its isolation from shallower, fresher waters. There was some discussion of the choice to have deep disposal for low-level and intermediate-level radioactive waste. In response, it was clarified that both near surface and deep disposal options were shown by studies to be capable of providing acceptable protection; the choice for deep disposal was made because it was the option favoured by the municipality and, thus, the decision was made through a more societal-driven process.

Development of a methodology for an environmental safety case in the United Kingdom

L.E.F. Bailey (United Kingdom Nirex Limited, United Kingdom)

This presentation introduced the notion of an environmental safety case that focuses on the protection of both humans and the environment. The UK safety case will address operational, transport and post-closure safety for intermediate-level waste (ILW) and high-level waste (HLW). As there is yet no agreed site or design for a deep geological repository in the UK, the first iteration of the safety case will be generic, drawing on examples from international repository concepts. These concepts will be considered in appropriate generic geological environments typical of those found in the UK. The performance of these example concepts will be assessed using a timeframes-based approach that focuses on the evolution of the multiple barriers and their associated safety functions. This approach recognises that the relative importance of the different barriers in providing safety will evolve over time. For example, at early times the engineered barriers provide containment and the geological barriers protect the engineered barriers and provide isolation of the wastes. At later times, as the engineered barriers degrade, the geosphere provides the major barrier to radionuclide migration back to the surface and ensures the long-term stability of the system. A multi-factor safety case will be presented, using multiple lines of reasoning, including comparisons with natural and anthropogenic analogues, to provide assurance of the intrinsic safety functions of the system and their evolution over time.

Strategy for safety case development: Impact of a volunteer approach to siting a Japanese HLW repository

K. Kitayama, K. Ishiguro, M. Takeuchi, H. Tsuchi, T. Kato, Y. Sakabe, and K. Wakasugi (Nuclear Waste Management Organisation of Japan, NUMO, Japan)

This presentation described the process for soliciting volunteer communities in Japan to explore siting an HLW repository. The intent in the Japanese approach is to start siting actions with volunteer communities only. A process timeline was illustrated and explained. Information is available on the internet at http://www.numo.or.jp/english. The presentation focused on one key element for assembling a safety case: namely, the management system and, in particular, the requirements management system with respect to different stages of repository site characterisation and development. In the first stage envisioned for the Japanese programme, preliminary investigation areas (PIAs) are selected from among volunteer communities based on literature surveys to confirm geological stability. The PIAs then undergo initial surface exploration as well as some characterisation through boreholes or geophysics to assess the feasibility and ease of making a safety case and, based on this, to identify sites as detailed investigation areas. Detailed investigation is conducted on the surface and in underground facilities and is aimed at supporting a provisional safety case with a comprehensive assessment of operational and post-closure safety to select a proposed repository site. After the presentation, the issue of communication with potential volunteer sites was identified as a challenge.

The Canadian nuclear regulatory process and use of the safety case for demonstrating the long-term safety of radioactive waste

P. Flavelle and M. Ben Belfadhel
(Canadian Nuclear Safety Commission, CNSC, Canada)

This presentation focused on the relation of the safety case to the Canadian process for licensing decisions on radioactive waste disposal and, in particular, on how the safety case can serve as a useful framework for managing information relevant to licensing. The Canadian Nuclear Safety Commission (CNSC) has its regulatory mandate defined by law. The burden of demonstrating safety is on the licensee, and the CNSC is responsible for ensuring that the licensee has met its responsibilities and that disposal does not lead to unreasonable risks to health and safety or to the environment. The safety case has several attributes that could be important to the licensing process, including providing a clear description of the disposal system and site, evaluating long-term safety, identifying key uncertainties, integrating complementary lines of evidence and arguments to build confidence, and demonstrating the commitment and expertise of the applicant. The licensing process takes account of this information as well as public input. A final decision is made by the commissioners of the CNSC. It was clarified that CNSC commissioners are political appointees, from qualified candidates provided by relevant agencies, who are independent of the CNSC professional staff.

The ONDRAF/NIRAS safety strategy for the disposal of B&C wastes

A. Dierckx, W. Cool, P. De Preter, P. Lalieux (ONDRAF/NIRAS, Belgium) and
P. Smith (Safety Assessment Management Limited, SAM Ltd., United Kingdom)

This presentation explained that the ONDRAF/NIRAS approach defines the safety strategy as the iterative process guiding the stepwise development of a geological repository and of its implementation procedures. The overall approach aims at developing a concept and design for the disposal of class B and class C (intermediate-level and high-level) radioactive waste in a geological repository, as well as procedures for repository implementation and measures to assemble evidence, arguments and analyses to show, through assessments, that disposal is both safe and feasible. The disposal concept provides a broad-brush description of the repository and its geological environment, along with describing the functions that they are intended to perform to protect the workforce during construction, operation and closure of the facility, and to protect the public and the environment in the longer term.

A sound safety strategy: (1) supports the development of the safety and feasibility case that is presented to authorities at key decision points, (2) provides a basis or platform for interactions with other interested stakeholders, (3) guides the research, development and demonstration activities (i.e. helps form a basis for research, development and demonstration plans) and (4) ensures that the decisions underlying the concept and design are well-founded, consistent and take account of various constraints or boundary conditions. Once made, such strategic choices are not expected to change much; for example, a decision that engineered barriers must ensure complete containment through the thermal period is a strategic choice. Such a strategic decision removes the need for thermal modelling to avoid complexity in both characterisation and modelling. Through a top-down approach, strategic choices are transformed into requirements. Then, through a bottom-up approach, the results of assessment and research-development and demonstration programmes permit confirmation of whether the requirements are fulfilled. As the safety case matures, instances of "should" in prior versions can become "does", as design and performance requirements are transformed by evidence into claims supporting safety.

Developing an advanced safety concept for an HLW repository in Rock Salt

J. Krone and N. Müller-Hoeppe (DBE Technology GmbH, Germany); W. Brewitz and J. Mönig (Gesellschaft für Anlagen- und Reaktorsicherheit mbH, GRS-Braunschweig, Germany), M. Wallner and J.R. Weber (Federal Institute for Geosciences and Natural Resources, BGR, Germany)

This presentation described the safety case concept being developed in the German programme for a proposed HLW repository in salt. Significant advances have been made compared to the performance assessment for salt developed earlier in the 1980s and mid-1990s. One new approach divides the repository into two main compartments for analysis: the engineered barrier system and the geological barrier. The integrity of the engineered barrier system is assessed based on functional requirements derived from the needs for containment (barrier tightness/durability) and for engineering design (structural and mechanical stability, limited crack evolution). The integrity of the geological barrier is assessed based on geological evolution processes and criteria, such as slow subrosion and small uplift potential. Concerning FEPs, the NEA FEPs database is considered as a starting point, but significant structuring and screening must be performed to use FEP information in the safety case. The observation was made that a challenge is always presented in performance assessment to justify assumptions, particularly for engineered barrier system performance. Some previous system-level safety evaluations focused on conservative scenarios and failed to emphasise the positive, expected performance. Emphasis is now shifting to describing nominal performance. Low probability failure scenarios will still be considered but not as part of the primary compliance demonstration.

Session III-b: The scientific and technical basis for assessing safety

Session III-b was chaired by K-J. Röhlig (GRS-Köln) and the rapporteur was F. Mompeán (NEA). Presentations in this subsession addressed: (1) defining the system concept, (2) describing scientific and technical information and understanding and (3) methods, models, computer codes and database.

Several papers in this sub-session addressed safety functions and design constraint requirements. Additional papers addressed the challenges from, and approaches for, managing, interpreting and integrating complex scientific and technical information. Other presentations illustrated approaches, methods, models, codes and databases to manage information and scenario development.

Six presentations were given as follows:

The phenomenological analysis of repository situations (PARS) – Application within the "Dossier 2005 Argile" (Meuse/Haute Marne site)

F. Plas and P. Landais (Andra, France)

This presentation explained the use of Phenomenological Analysis of Repository Situations (PARS) in Andra's "Dossier 2005 Argile" to identify and characterise phenomena in repository evolution (out to 1 million years into the future). Under Andra's approach, the evolution of the repository is semented in space and time into "situations," which represent the phenomenological state (in terms of relevant processes) of part of the repository or of its environment during a given period of time. The ability to "segment" the disposal system in this way is facilitated by a modular design, by the layered structure of the geological medium, by identifying certain "triggers" events for key events or processes, and by defining various timescales which characterise important phenomena. A number of "situations" can then be compiled into a "scenario" which can be imagined to be like a movie that represents an evolutionary path of situations.

PARS is being used for analysis of numerous repository situations including these examples:

- Hydraulic and gas transients for various cells and drifts
- chemical and mechanical evolution of the components within waste cells
- Thermal-mechanical transients for cells with spent fuel.

The use of PARS in "Dossier 2005 Argile" provided a comprehensive view of the main types of phenomena and their relative importance, improved understanding of the interactions among phenomena, and assisted in the identification of major uncertainties.

In the discussion following this presentation, the speaker affirmed that the PARS output will feed information to performance analyses supporting the next design stage. It was also mentioned that, for Andra, the PARS approach has been not only a technical tool but also a organisational and management asset, in that applying the method has effectively fostered team development.

Knowledge management: the Cornerstone of a 21st century safety case

H. Umeki, H. Osawa, M. Naito, K. Nakano, and H. Makino (Japan Atomic Energy Agency, JAEA, Japan); I.G. McKinley (McKinley Consulting, Switzerland)

This presentation put forward ideas regarding knowledge management and its role in a safety case. The observation was made that no single person can understand all aspects of a given discipline, let alone the multiple disciplines involved in the evaluation of long-term repository behaviour. Can a "radwaste Leonardo da Vinci" be redefined for the 21st century who can cope with the exponential growth of information? This is both a challenge and a fertile area for international collaboration (this suggestion was supported by a comment from the audience). The safety case can be used to structure knowledge: knowledge creation, manipulation and the need for new knowledge can be identified in the safety case, and would change in scope and emphasis as programmes mature, knowledge levels change and other information needs are identified.

The approach of the Japan Atomic Energy Agency (JAEA) to knowledge management was explained in some detail. Three main elements of the knowledge management system (KMS) that is being built at JAEA were illustrated:

- Training (development and implementation challenges).
- Electronic assistant (to be developed in steps and change in response to needs; to take and automate all roles except synthesis and co-ordination).
- Think tank (*inter alia*, to anticipate future requirements).

KMS will keep a full record of knowledge meaningful to defining, understanding and evaluating the repository (a legal requirement). This is not seen as a burden but as a challenge that will, incidentally, allow much information-technology development to take place. It is a tool for decision making, and it will change in time as the relative importance of pieces of information may change in time.

Safety functions and safety function indicators – Key elements in SKB's methodology for assessing long-term safety of a KBS-3 repository

A. Hedin (Swedish Nuclear Fuel and Waste Management Company, SKB, Sweden)

This presentation introduced the topics of safety functions and safety function indicators (SFIs). SKB has used these in the safety assessment SR-CAN, whose main purpose is to provide feedback to design development, research and development programmes, site investigations, and future safety assessments that might be used to support license applications. In the SR-CAN assessment, SKB defined safety functions and safety function indicators. A safety function embodies the key roles of a repository component in terms of its contributions to safety of the disposal system (e.g. the canister should withstand isostatic load). Based on the safety functions, "safety function indicators" can be defined to quantify performance; a safety function indicator (SFI) is a measurable or calculable property of a repository component that indicates the extent to which a safety function is fulfilled. (e.g. isostatic stress in a canister). Criteria can then be developed to establish quantitative limits (or thresholds) by which one can judge whether a safety function is being fulfilled (e.g. isostatic stress must be less than isostatic collapse load.). In accordance with the KBS-3 disposal concept, safety function indicators in SR-CAN related primarily to the near-field.

These safety function indicators are related to, but not the same as, design criteria; design criteria relate to the initial state of the repository, whereas SFI criteria should be fulfilled throughout the assessment period (1 million years). Design criteria should be established so that, ideally, taking into account evolution and deterioration of system components, all SFI criteria are fulfilled throughout the full assessment period. It was emphasised that breaching SFI criteria does not necessarily imply an unsafe repository; these indicators are complements to, not substitutes for, the system-level safety measures of interest (i.e. radiological dose and risk). It may be, for example, that SFI criteria could be defined that would allow certain processes to be completely excluded from consideration; safety could still be assured if the criterion is breached so long as the process is evaluated quantitatively and the consequences are not sufficient to endanger additional safety functions. It was noted that several aspects of repository evolution cannot be easily captured by a simple comparison to an SFI criterion.

SKB found the safety function approach to be very useful in helping focus on the critical issues in safety assessment and by providing a structure for defining and evaluating the a comprehensive main scenario as well as in selecting additional scenarios for assessment. In discussion, the relationship between functional and FEP-based assessment approaches was addressed. It was suggested that SFIs are instruments to discuss and prioritise FEPs once a project is mature. Additionally, it was suggested that it would be good to have a pre-agreement with stakeholders on SFIs and safety function criteria so that they do not become misused or misinterpreted, and so that there is a good understanding that safety is not necessarily compromised if a criterion is not met.

The role of structural reliability of geotechnical barriers of a HLW/SF repository in Rock Salt within the safety case

N. Müller-Hoeppe (DBE Technology GmbH)

In this presentation, the focus was the role of structural reliability in a safety case, specifically that of geotechnical barriers of an HLW spent fuel repository in salt. Technical standards for civil engineering were a starting point for determining structural reliability needs. Actions (on the geotechnical barrier) were defined in terms of thermal-chemical-hydrological-mechanical loads, and resistances were defined to be capacities that could withstand representative combinations of actions. In the context of technical standards, two methods are distinguished: the calculation method and the design (assisted by testing) method. Proving structural reliability requires calculations that address safety criteria, which require validated models and uncertainty management. When applying the design/testing method, the portability of the test conditions must be guaranteed.

An illustrative case in point was the structural resistivity of the drift seals of the Morsleben low-level radioactive waste (LLW) repository (ERAM) in Saxony-Anhalt, Germany. Its proof of structural resistivity is mainly based on the calculation method; calculations are still being performed to ensure that relevant features are captured. The flow resistance of the ERAM drift seals was evaluated using the design/testing method. Regarding the Salzdetfurth shaft seals, structural resistivity as well as the flow resistance were determined by testing.

A leap had to be made from the standard industrial technical standards lifetime of 100 years to the radioactive waste programme requirements for a design life of 5 000 to 30 000 years. Regarding the ERAM drift seals proof of durability over these timeframes was accomplished by extrapolation of experimental results on alteration of salt concrete.

Understanding the evolution of the repository and the Olkiluoto site

K. Koskinen (Posiva Oy, Finland) and B. Pastina (Saanio & Riekkola Oy, Finland)

This presentation provided an introduction to the evolution of a KBS-3V type repository at the Olkiluoto site. The "Evolution Report" was described as a key element of the safety case using a process-based approach. It acknowledges that there is going to be a knowledge deficit, meaning that there is currently no way to ensure whether new processes may appear when propagating known processes in time and space. Two climate scenarios (one with and one without human-induced climate effects) with a defective waste package variant were described for the post-closure phase. A main finding from the current evolution report is that the copper canister corrosion depth, up to 100 000 years after closure, is expected to be only a few millimetres out of the 50-mm canister thickness. The evolution report is to be updated in 2009 (as part of the preliminary safety analysis report) and 2011 (as a stand-alone report).

Issues to be addressed in future updates include:

- Evolution of the buffer and backfill.
- Flow paths to and from the repository.
- Excavation-damaged zone effects.
- Cementitious materials effects on engineered barrier systems.
- Glaciation effects on engineered barrier systems and host rock.
- Repository closing and sealing issues.

Making the post-closure safety case for the proposed Yucca Mountain repository

P. Swift (Sandia National Laboratories, SNL, United States of America) and
A. Van Luik (Department of Energy, U.S. DOE, United States of America)

This presentation provided an overview of the Yucca Mountain repository post-closure safety case. The safety case concept is being integrated into the license application being prepared for Yucca Mountain, by giving particularly close attention to the treatment of uncertainties, thereby bringing available lines of evidence into the supporting information, as appropriate, to build a comprehensive argument for safety and regulatory compliance. For Yucca Mountain, it is expected that there will be open questions in the safety case to be presented to the regulator and a programme will be outlined on what information is to be gathered (and how) prior to the next iteration in the licensing process to address such open issues. A one-hundred year operational phase is foreseen and planned, and the changes in knowledge and approaches that occur over time will have to be accommodated through the formal licensing process.

It was noted that the Yucca Mountain geological system differs significantly from other systems being considered among participants (unsaturated fractured rock and an oxidising environment); nevertheless, an area of common interest highlighted in the discussion was the role of institutional actions in building confidence. Some institutional actions related to Yucca Mountain that are deemed important in this regard are a well defined programme of testing to challenge data and models during the operational phase, providing clear explanations of planned pre-closure and post-closure controls, supporting the quality assurance programme, and the institution of a "safety conscious work environment" (meaning any worker is expected to challenge any condition or action that could potentially degrade safety).

Session III-c: Evaluating and documenting confidence in safety

Session III-c was chaired by H. Umeki (JAEA) and the rapporteur was J. Andersson (JA Streamflow AB). Presentations in this subsession were focused on the following:

- Statement of confidence.
- Evaluating uncertainty in the context of confidence.
- Compliance measures and safety indicators.
- Quality plan and peer reviews (internal and external).

Summaries of the presentations are provided as follows:

Elements of the safety case for the Morsleben repository based on probabilistic modeling

J. Wollrath (German Federal Office for Radiation Protection, BfS, Germany); M. Niemeyer,
G. Resele (Colenco Power Engineering AG, Switzerland); D-A. Becker and P. Hirsekorn (Gesellschaft für Anlagen- und Reaktorsicherheit mbH, GRS, Germany)

This presentation provided an overview of the safety case and probabilistic modelling elements for the Morsleben LLW repository. The Morsleben repository (ERAM) in Saxony-Anhalt, Germany, is located in a former salt mine. Operation started in 1971 and ended in 1998. The licensing procedure for the closure of the repository has been initiated. A safety assessment is part of this license application.

The closure concept is based on extensive backfilling of the salt mine with an inexpensive concrete mixture (salt concrete). Several potential closure concepts were elaborated and analysed with respect to technical feasibility and long-term safety, and were assessed by different teams. The main idea of the closure concept finally selected is that the entire system exhibits a barrier effect through a partially redundant combination of several processes. In particular, the extensively backfilled mine has a high resistance to movement of brine. Processes to be considered in accordance with the safety concept and its different components were identified. These processes form the basis for modelling the system.

Safety assessments for ERAM have been performed independently by two groups using different conceptual models and codes, and with, as much as possible, identical parameters. Calculated radiation exposures are below applicable dose constraints. The results of the two models correspond satisfactorily and the results from probabilistic calculations corroborate those from deterministic calculations. Differences in the results from the two independent groups are well explainable and are related to the different model conceptualisations. These findings significantly enhance the confidence in modelling results and contribute significantly to the safety case for the geological disposal of radioactive waste at the Morsleben site.

In spite of the simplifications of geometry and the processes involved, more than 200 parameters are necessary to describe the system. Some of these parameters are well defined but most of the parameter values cannot be characterised exactly and are therefore considered with a large bandwidth. The system is highly non-linear. Probabilistic analyses show that more than 99% of the realisations are well below the dose constraint of 0.3 mSv/a, with few remaining realisations below 1 mSv/a. Detailed analysis shows that the few realisations that slightly exceed the dose constraint of 0.3 mSv/a result from a combination of parameter values lying close to the extremes of their bandwidths; a less conservative choice of the parameter distributions or of model assumptions would clearly avoid even these few calculated realisations.

As to the challenging of co-ordinating and organising, the work of multiple independent groups, ERAM acknowledged that this was a learning experience that took several years (and iterations) to work smoothly, but that it was worthwhile.

European pilot study on the regulatory review of the safety case for geological disposal of radioactive waste

F. Besnus (Radiological Protection and Nuclear Safety Institute, IRSN, France), J. Vigfusson (Federal Nuclear Safety Inspectorate, HSK, Switzerland), R. Smith (U.K. Environment Agency, United Kingdom), V. Nys (Association Vinçotte Nucléaire, AVN, Belgium), G. Bruno (European Commission, EC), P. Metcalf (International Atomic Energy Agency, IAEA, Austria), C. Ruíz López (Nuclear Safety Council, CSN, Spain), E. Ruokola (Radiation and Nuclear Safety Authority, STUK, Finland), M. Jensen (Radiation Protection Authority, SSI, Sweden), K. Röhlig (Gesellschaft für Anlagen- und Reaktorsicherheit mbH, GRS-Köln, Germany) and P. Bodenez (Nuclear Safety Authority, ASN, France)

This presentation described the European pilot study on regulatory review of the safety case. A number of countries within Europe are developing or giving consideration to the development of geological disposal facilities for HLW. The safety authorities in Belgium, France, Finland, Germany, Spain, Sweden, Switzerland, and U.K. are interested in exploring the possibility of a harmonised approach to the demonstration of safety of such facilities and the regulatory review of documentation providing such demonstration. As such, and with technical support organisations and international bodies, they have initiated a pilot study on how these elements should be presented in a safety case, for, *inter alia*, regulatory review and approval.

Key stages of a repository project are: conceptualisation, siting, design, excavation, construction, operation and closure. Four issues have to be addressed at each stage: facility design and safety strategy, demonstration and site-and-engineering suitability, impact assessment and adequacy of management systems. The impact assessment is not solely sufficient for a safety case.

Uncertainty and its management have been examined in detail within the pilot study, specifically to identify commonalities and differences in approach and, thereby, strengthen guidance in this area. The focus on uncertainty management was made because it plays a major role in the interplay between science, design and safety study. It is central when developing a repository system and assessing its safety.

It has been found that the regulatory frameworks differ considerably between countries, but the regulatory attitudes towards the achievements and demonstration of safety differ to a lesser extent. The importance of the stepwise approach is acknowledged. Furthermore, it is essential to keep the regulatory/licensing authority and technical support organisations informed at each step, and involve them in decisions even if it is not formally required. Regarding management of uncertainties, it is essential that implementers present uncertainties and clear strategies for resolving uncertainties. Regarding compliance, dose/risk are only broadly conservative indicators – disaggregated results are needed. There is ongoing discussion within the study regarding timescales.

Several questions were discussed. The first was whether, in examining the treatment and management of uncertainties, the study had tried to address the issue of "the unknown unknowable" – i.e. especially, events or processes whose possibility of occurrence or change is essentially impossible to quantify or confirm, especially over the timeframes of geological disposal. It was noted that this issue had been discussed within the group, but not addressed in the report. The view was expressed that implementers should be limited to assessing the range of scenarios that covers the range of behaviours of the safety functions.

Regarding the study conclusion that differences between regulations and regulators are not significant, the question was asked whether this result might be due to the level of analysis. That is, were differences in more detail like requirements on quality assurance or elicitation of expert judgement taken into account? The response emphasised that the purpose of the study was to reach common views concerning regulatory practices, if possible, rather than to focus on differences. There was agreement on the most important provisions and, even when regulations differ, in practice there were many similarities in implementation and interpretation. For example, regulations on reactor safety differ within Europe but there is a common view and framework. The study group that particularly examined differences between the organisations regarding uncertainty management did not find any fundamental disagreements. Finally, it was clarified that the study on uncertainty management focused on post-closure safety and did not attempt to address operational safety.

From initial safety considerations to a final safety case: Forms and purposes of safety analysis during a disposal project's lifetime

J. Vigfusson (Swiss Federal Nuclear Safety Inspectorate, HSK, Switzerland)

This presentation provided an overview of the ongoing safety analysis in the evolution of a safety case for a geologic repository. Repository development is a long-term project. During preparation and realisation of a radioactive waste disposal project, a multitude of decisions are made, which require input from an assessment of the radiological safety of the disposal facility. The rationale for the decisions must be documented to inform decision makers in coming steps. Depending on the stage of the project and the purpose of the assessment, the safety assessments that are needed – or are possible – vary in comprehensiveness and in the choice of safety indicators. The scopes of such different safety cases are currently being assessed by the Swiss regulator and thoughts are still developing.

The earliest safety case, the "Generic Safety Evaluation", serves a role at the beginning of the site selection process. It is not site specific. The implementer must declare to the regulator what type of repository is being considered. Based on the choice, the Generic Safety Evaluation illustrates the necessary isolation time of the waste and shows the contributions to confinement and isolation expected by engineered barriers. From this information, the implementer must draw quantitative conclusions on the levels of safety-relevant qualities required of the host and regional geology.

Preliminary safety cases are developed during the second stage of the site selection process. They are site specific. They serve to support decisions on whether to consider the contributions to safety offered by different sites as comparable. Best estimates for parameter values are not sufficient; different levels of uncertainty need to be considered in the comparison.

The safety case for the general license application is needed to support the decision to enter the licensing process with a selected site, which is the final and most important milestone in the site selection process. This decision must be well founded, having followed a scientifically and politically sound selection process based on available information to support a thorough analysis of the final candidate site.

The safety case will also evolve during the lifetime of the repository. Safety cases are repeatedly needed at major decision points during the project lifetime. The focus of each safety case changes according to the purpose served. During the project history, the refinement of the safety cases will increase as information gradually becomes more complete and precise, and thanks to the development of better analysis tools. But there will be other changes too: the audiences will change, the general scientific background will evolve and so will the philosophical attitudes in society. The stepwise development of disposal facilities, with the possibility of independent reviews and public involvement at each step, allows us to adjust and improve projects as such developments unfold.

There was extensive discussion on this presentation: it was suggested that during site selection, it is not always clear how to decide between sites and options, as many different factors need to be balanced; not only technical factors but also in terms of public acceptance. The objective is protection and selection of the "best" site does not serve that objective if a disposal system at that location can never be implemented. The presenter agreed that this was not an exercise aimed at comparing "the numbers behind the comma." Instead, safety classes will be compared, probably accepting a range of two orders of magnitude within each class. Furthermore, in the Swiss programme, the view is that there is plenty of time, thanks to sufficient interim storage capacity and adequate financial resources; short-term political considerations will not be allowed to get in the way of building a safe repository. Another commenter observed that there is a time aspect to safety; the longer it takes to realise a repository, the higher the risk. A final question addressed whether there was not a risk of raising expectations by suggesting one could use safety cases to distinguish between numerous sites that are suitable. In addition, it was argued that the ease of making the safety case for a given site may be more important than the calculated consequences. In sum, if it is uncertain whether the analysis is possible, then the site has to be considered less useable. In that case, sites with more information or fewer uncertainties may be preferable so long as they provide equivalent protection – but as noted earlier, this "equivalency" can be judged within a rather large range.

Data gathering and long-term maintenance of confidence

Z. Nagy (Public Agency for Radioactive Waste Management, PURAM, Hungary)

This presentation focused on maintaining confidence in the safety case and the important role of data gathering. According to Hungarian legislation, safety assessment is required in each licensing phase of repository development: preparation, establishment, operation, and closure. During the operation of the repository, the safety assessment should be reviewed regularly and, therefore, the safety assessment will be made many times during the repository development process.

Between two consecutive safety assessments, many decades may elapse; between the first and the last safety assessments, more than one century may elapse. The assessment of long-term safety is based partly upon geoscientific information. It is not yet clear for how long historical data, originating decades beforehand, may continue to be used. In addition, people who participate in the decision-making and safety-assessment process also change and the decision-assessment environment will change over time. Thus, there is a challenge to maintain over many decades the confidence in geoscientific information gathered now. If such confidence cannot be achieved, then older data may be excluded from use in later safety cases and could thereby reduce the assessment capability by reducing the amount of available data.

To support continued confidence and to avoid the need to assemble new data, it might be possible to store sufficient metadata to allow for a quality assurance reassessment of the old data. To accomplish this, it is necessary to define what additional metadata must then be integrated into multidisciplinary geoscientific databases for this purpose. Databases must contain not only the observed information but as much metadata as possible (i.e. information about the condition under which the data was gathered and the interpretation methods that were used).

In the discussion, it was observed that although this presentation focused on site data, the issue is relevant to all information in the safety case. It would be useful to compile an international guide about the interaction between data gathering and long-term maintenance of confidence.

Focus on isolation and containment rather than on potential hazard: An approach to regulatory compliance for the post-closure phase

B. Baltes, A. Becker, A. Kindt and K.-J. Röhlig (GRS-Köln, Germany)

This presentation described a regulatory approach focused on isolation and containment of waste rather than on potential hazard and dose. In recent years, consequence calculations have been put in perspective in relation to other evidence used in a safety case. This development has been considered good and necessary, while recognising that such calculations still play an important role in a safety case. There is no single recognised method to judge the regulatory compliance of calculation results in the regulatory process. Traditional comparison to radiological criteria is still an element, but questions arise. How reliable are the results? How can we know that the environment is protected? And so forth. The present approach for the ongoing revision of safety criteria in Germany is based on demonstration of the confinement of radionuclides.

The Committee on a Selection Procedure for Repository Sites (AkEnd) introduced the concept of the "isolating rock zone", which comprises a portion of the geological environment that immediately surrounds the repository and that can maintain crucial properties over a timeframe of 10^6 years. This concept has led to a regulatory approach with safety criteria that focus on the geological barrier.

The evolving German approach is to assess the "completeness of confinement" (i.e. for likely scenarios, the releases should not cause significant increase over consequences from natural sources). This means that calculated radionuclide concentrations in the accessible environment are seen as primary safety indicators, which allow, together with other lines of evidence, judgment about the confinement capability of a repository system. Assessment calculations are just one (important) input to the safety case. Dose is not seen as an indication of hazard to a hypothetical individual, but – together with other suggested indicators (six were described) – as a means to judge calculated concentrations. This work is still in progress and there are several open issues, including implementation of a stepwise approach and understanding the role of those indicators for which the predictability is limited in time.

There was considerable discussion following this presentation. It was reiterated that interpreting the meaning of traditional radiation protection criteria in the geological disposal over very long time frames is fraught with challenges. Given that dose is still retained (instead of being replaced) as an indicator in the German approach, the question was raised whether formulating additional criteria might add complexity and uncertainty rather than reducing it. In response, the presenter noted that it is difficult to eliminate dose as an indicator; it is a familiar measurement quantity, and there is a desire by regulators and stakeholders to see dose calculations, even if they cannot be viewed as accurate indicators of performance far into the future. The use of additional criteria provides insights and allows direct evaluation of factors that relate more immediately to the geological environment and, thus, to aspects of the disposal system that are much less subject to uncertainty – and, at the same time, more critical to safety – than those necessary for traditional dose assessments (e.g. surface aquifers and biosphere).

Another alternative approach suggested was to consider applying the concept of best available technology with a focus on preventing releases and not as much on lowering calculated dose consequences. Finally, it was observed that this approach could work well for salt or clay host environments, where confinement by the near field is a prominent feature. It may not be readily applicable to granitic formations where greater reliance is placed on containment by the engineered barrier systems. Nevertheless, this approach could possibly be applied for other disposal systems, taking into account the properties of the relevant host rock.

The UK Environment Agency's assessment of BNGSL's 2002 post-closure safety case for the low-level radioactive waste repository at Drigg

A.J. Baker, D. Bennett, S.L. Duerden and I.J. Streatfield
(Environment Agency of England and Wales, United Kingdom)

This presentation focused on the regulatory assessment process of the post-closure safety case for the LLW repository at Drigg in the United Kingdom. The Environment Agency of England and Wales (The Environment Agency) regulates radioactive waste disposal in accordance with the Radioactive Substances Act 1993 (RSA93). British Nuclear Group Sellafield Ltd. (BNGSL) is currently authorised to dispose of solid LLW at a repository near the village of Drigg, close to the Cumbrian coast in North West England. In accordance with government policy, The Environment Agency reviews authorisations for the disposal of radioactive waste periodically. The Environment Agency assessed BGNSL's 2002 post-closure safety case for the LLW repository to inform its recent review of the Authorisation.

The 2002 post-closure safety case considers the long-term impacts of the facility on the environment and on the public after waste emplacement operations have ceased, and the safety case includes a post-closure radiological safety assessment.

The 2002 safety case consisted of a top-level document, but relied on more than 200 supporting documents. The latter contained significant information and evidence supporting the assessment, but many of the documents were made available several months after the submission of the safety case. The timeframe opened questions about the relative role of the different documents submitted. The Environment Agency decided on having a limited exchange with the operator during its review of the safety case to ensure that the safety case stands on its own merits.

Regarding confidence in the review, several lessons were learnt. Building confidence in judgments was helped by various means, including contributions from a wide range of scientific specialists, risk assessment to focus the review effort, and detailed audits of selected parts of the implementer's work. The Environment Agency has established a clear documented approach and criteria for the review. A wide range of experts who are independent from the industry were used. It was noted, however, that as programmes move toward licensing, it can be increasingly difficult to engage experts who have no connections with the implementer. A clear audit trail for review findings, and decisions based on those, was established. Consideration of the most appropriate means for communicating findings from the review is essential, as it is a challenge to adjust the communication to different audiences. Early dialogue between the implementer and the regulators is recommended. A balanced design of the safety case is required that builds a safety case rather than simply links individual calculations and scientific studies. This requires understanding and presenting the key safety issues and safety arguments backed up with quantitative and qualitative indicators and audit of uncertainties and biases. Whilst the review was specifically focused on an existing disposal facility for LLW, the Environment Agency considers that its experience has highlighted some important issues that might be considered when developing and implementing a process to review a safety case for a new, deep geological disposal facility.

For the future, more guidance on the content of the safety case is needed. Issues of concern are: long-term management of information, time periods, optimisation, human intrusion, supply of information, confidentiality, dialogue during the review and making the review more publicly transparent.

In discussion, it was noted that experience with NEA peer reviews shows dialogue during the review is essential to prevent misunderstanding and to clarify certain issues. In addition, it is a common problem to get new information during a review. Late-coming information from the implementer can be important and is often welcomed by reviewers even when it is late in the review cycle.

Perspectives on developing independent performance assessment capability to support regulatory reviews of the safety case

J. Winterle (Center for Nuclear Waste Regulatory Analyses, CNWRA, United States of America) and A. Campbell (Nuclear Regulatory Commission, NRC, United States of America)

This presentation addressed the role of independent performance assessment capability for a safety case from a regulatory perspective. The U.S. Nuclear Regulatory Commission (NRC) has responsibility under U.S. statutes and regulations to conduct a review of the DOE license application for a repository at Yucca Mountain, Nevada, in the U.S. A key component of the DOE license application will be a total system performance assessment to demonstrate compliance with regulatory requirements for post-closure performance. The NRC, with assistance from the Center for Nuclear Regulatory Waste Analysis (CNWRA), have developed an independent total system performance assessment model of the potential Yucca Mountain repository, which provides a tool for assessing the significance of various factors to overall repository performance and regulatory compliance. For FEPs in the Yucca Mountain performance assessment, the level of emphasis in an NRC review will be based on whether it is identified as a barrier by the DOE and by the relative significance inferred from independent analyses by the NRC.

It was observed that there are also some drawbacks to regulators developing independent performance assessment capacity, the main issue being cost. As an alternative, regulators may instead decide to assess critical subsystems and leave the total system modelling to the implementer. It was also noted that there are differences between the NRC model and the DOE model, given their independent development and purposes. It is not the objective of the NRC to make a safety case, but rather to understand the importance of different processes. This allows for substantial simplification of the sub-models that the implementer needs to apply in full. Furthermore, the regulator may chose to include FEPs that are not being considered by the implementer, but the regulator needs to be clear that the implementer is not required to include such FEPs.

The discussion affirmed that it is extremely important to a project that there is independent capability on the regulatory side. This should not be forgotten even if it can be challenging in terms of budget or time constraints. The presenter was asked if his organisation aimed for a complete or a partial system performance assessment capability. The response was that to some extent it is partial, because there is a need to rely on what the implementer provides in terms of data. The interpretation of that data, however, is still made independently.

Session IV: The embedding of the safety case in societal dialogue and decision making

Session IV was chaired by Alan Hooper (United Kingdom Nirex Limited, United Kingdom) and the rapporteur was Bo Stromberg (SKI). Summaries of the presentations are provided as follows:

The partnership experience on disposal of low- and intermediate level short-lived waste in Belgium

P. De Preter, W. Cool, E. Hooft, A. Waffelaert (ONDRAF/NIRAS); J. Blommaert and J. Draulans (Study and Consultation Radioactive Waste Dessel, STORA Dessel)

There has been a successful partnership between ONDRAF/NIRAS and Belgian municipalities in gaining acceptance for a low-level and intermediate-level short-lived radioactive waste disposal site. The municipalities in Dessel and Mol agreed to siting after a six-year-long pre-project consultation programme. The success was based on a well-defined decision-making process with broad participation on the local level. It was important for the process that the municipalities received funding both for staff members and external independent experts. Initially, a large effort was devoted to information exchange and knowledge-building within four separate working groups. Subsequently, a study-and-evaluation phase was initiated; the process was ended with conclusive discussions about the repository. Confidence in repository safety was mainly achieved through an understanding of the intrinsic qualities of the disposal system, rather than by dose calculations.

During discussions, it was noted that, for the process in question, regulators participated only a few times in the consultations with the local representatives. The importance of an independent regulator was recognised, but it was emphasised that the implementer had the ultimate responsibility for the consultations.

In terms of organisational issues for the implementer, it was acknowledged that, in the beginning, there was reluctance towards an engagement in the consultation process among some staff members and there was a tendency to be defensive in public dialogue. It was a learning process during which it became apparent that not all experts have the talent to discuss technical issues in a public venue. With regard to funding, the consultation was based on an annual budget which also included extra money for external independent experts. Finally, it was clarified that the community partnership decided on the issues that should be brought up in discussions with the implementer these issues were subsequently discussed with the regulator, thus lowering the risk of posing a conflict with regulatory priorities.

A site county's perspective on oversight activities and the safety case:
Nye County (Nevada, USA) local perspective of the Yucca Mountain project

D. Swanson (Nye County Nuclear Waste Repository Project Office, United States of America)

Nye County (Nevada, USA) is host to the potential Yucca Mountain repository. The county has a longstanding relationship to the nuclear sector as host to numerous nuclear explosions in the U.S. nuclear weapons development programme. It is a huge county in terms of land area, but not in terms of population, which is currently around 44 000 people. Nye County has no veto power related to the Yucca Mountain programme, but it does have a constructive approach to the project. The most important objectives of Nye County are safety and health protection. Economic benefits and a successful repository programme are also important. Nye County would like the DOE to become more involved in regards to further development of local business and residential land, amongst other community interests. A transportation railway for nuclear waste is considered an important component that should, ideally, be built as soon as possible; truck transportation is viewed as a less favourable and a less safe option. Nye County has established the Nuclear Waste Repository Project Office expressly for activities related to the Yucca Mountain Project. Important objectives of the office are achieving dialogue with the DOE, participating in DOE review activities, developing requests for impact assessments, and engaging in monitoring and testing studies. The oversight programme is used to inform the residents of both Nye County and the State of Nevada about the Yucca Mountain Project. The monitoring work is mainly related to geology, hydrogeology and engineering.

A questioner asked whether Nye County had commented on the new EPA proposed regulations for long-term safety of Yucca Mountain. The response was affirmative, but the speaker stated that in Nye County, long-time perspectives beyond 10 000 years are of lesser concern. There is already significant radioactive contamination from previous activities at the Nevada Test Site that is locally considered a more serious problem.

Sweden: Role of stakeholder review of technical papers
(Oskarshamn LKO project case study)

K. Nilsson (Local Competence Building in Oskarshamn Municipality –
Nuclear Waste Project, LKO, Sweden).

The Oskarshamn Municipality is currently hosting site investigations (one of two candidate sites) carried out by the Swedish Nuclear Fuel and Waste Management Company (SKB) with the aim to locate a repository for spent nuclear fuel in Sweden. The municipality is already host for three nuclear reactors, the interim storage facility for spent nuclear waste (Clab) and the Äspö Hard Rock Laboratory. SKB has recently submitted an application to build an encapsulation plant near the existing Clab facility. There is a large majority in favour of the SKB programme in the municipality. The key issue for the inhabitants is whether a repository will be safe. To come to such a conclusion, there needs to be trust in the safety case. Such trust needs to be gradually built up through an involvement, dialogue and agreement between experts and laypersons. There needs to be trust both in the implementer and the regulator.

Cornerstones for success include: (1) a clear description of the project, (2) a defined decision-making process (rules must be established early and must be followed) and (3) an open process where uncertainties and problems are acknowledged and discussed. There needs to be distinction between facts and values, because the values of laypersons, environmental groups, and other interested parties can be just as well-founded as those of the implementer. In Oskarshamn, the community oversight organisation (the LKO project, focused on "local competence-building" on the topic of nuclear waste) is funded by the nuclear waste fund (700 000 Euros per year) and the local activities are focused around three working groups that hold monthly meetings. Work within the safety and neighbour groups includes critical reviews, whereas the future perspective group focuses on the development of the municipality.

Discussion was held regarding whether there was awareness (by the municipality) of the NEA guidelines for the safety case and this topic was suggested for future work within the NEA Forum on Stakeholder Confidence. On the local level there appears to be support for a repository, but public support appears to diminish with distance from the proposed repository site. In the Swedish case, good support is shown in both municipalities in which site investigations are taking place (Oskarshamn and Osthammar), and also in the Hultsfred municipality next to Oskarshamn where SKB has previously carried out a feasibility study.

Some symposium attendees appreciated a car-dealer analogue regarding how a similarly persuasive "seller" approach can deter public opinion. Attendees also liked the attitude expressed that the public is a resource. It was recommended that messages to laypersons be kept simple and well presented to promote a comprehensive understanding of the main conclusions of a safety case. SKB has managed to do this very well.

Session IV Panel Discussion

Professor van Hove opened the panel session by observing that organisations are slow to change when it comes to the elicitation of stakeholder co-operation on the subject of the safety case. Yet, achieving and maintaining stakeholder trust is to be sought not only on specific issues as they arise, but as a constant endeavour. Seeking stakeholder trust ought to be organic to the waste management institution: "It is a daily job, years on end". Professor van Hove then introduced three thought-provoking observations that are often put forth by implementers or regulators, but that lay stakeholders may find not credible:

1. Statements of continued long-term integrity and safety of the repository: The historical experience is that man builds for perpetuity – roads, cemeteries, and so forth – but none of these projects will survive forever. Pyramids, even though they have persisted for millennia, were looted relatively quickly and failed to perform their primary function.
2. Statements that a facility will fulfil its intended purpose without need for maintenance: "Anything done by man needs minimum attention" to not degrade over time.
3. The treatment of risk: people are willing and sometimes eager to assume risks as long as they are in control, but they do not want risk imposed upon them. Involving stakeholders ought to ensure understanding and clarity of the risks that they would be taking and ought to sufficiently empower them to do so freely.

The audience was responsive to Professor van Hove and the following discussion ensued:

- The demand for a certain level of performance is the demand of regulatory organisations that require a technical response. The regulations may or may not represent the demands and concerns of stakeholders. The public is familiar with the concept of "generations", rather than "centuries", and the public voices definite requests for safety during, say, the next 100 years. Public discourse appears hazy when a few centuries are being considered; when longer than 1 000 years, it becomes "a matter of science". The public generally finds assurance in knowing that serious studies have been performed and that radioactivity is contained. Technical requests, however, may come from the public because they observe geologic changes on the surface of the earth and the claim is made by repository proponents that the timescales for deep underground geologic changes are longer than what they observe at the surface (e.g. erosion and earthquake damage). Geologists are accustomed to working with timescales of millions of years whereby the stability of the rock mass can be predicted to exist, even if the stability of engineered barriers becomes questionable.

- The notion of stability in time has a symbolic component. To build "forever" is often interpreted to mean building as soundly as possible and then adapting "as the need arises". This capability exists as when the results of an experiment show a necessity to benefit mankind or the environment. There is also a large degree of symbolism in the way engineering projects are presented. Regarding issues of long-term maintenance, for instance, concrete is often conceived to have more longevity than do bricks, but both bricks and concrete withstand the test of time. Such symbolic imagery resides in perception.
- There are roles in the safety case for both scientists and philosophers. The discipline of developing and testing the safety case clearly results in enhanced safety. The example of the car industry is pertinent: cars are tested in labs and certain tests can only be done in the lab, not on the road, and yet safer cars are produced. Science is built on "certainty for the moment", based on research. The philosopher acknowledges the scientific certainty for the moment, rather than the absolute value, and recognises that the role for calculations and regulations is to serve for comparison. The statement "safe for 100 000 years" relates to a set of calculations and tests under a set (or range) of assumed or postulated conditions. It is not a prediction of reality, but can serve as a standard that allows a design or project to be judged against other projects, courses of action, or risks.
- Decisions regarding safety are both political and technical. Decisions based on scientific analyses often depend on politically-correct timing so that relevant circles are comfortable with a certain path at a point in time. Such decisions must be prepared and, to some extent, must pre-exist in the political context of the moment. It is beneficial to both implementers and stakeholders to be cooperative and to mutually agree on paths forward, thereby increasing the possibility of informed political decision making.
- Considering that risk is integral to everyday life, public adversity is often encountered when confronted by unsolicited imposed risk. Reducing imposed risk is often achieved in hosting communities through participation in planning and development of a geological repository. This participation allows the community to become more knowledgeable and accepting of potential risks associated with a geological repository than people from outside the community. In other words, the positive attitude towards a repository often decreases with the distance from the proposed repository location.
- Public dialogue, negotiating with communities and co-deciding is more encompassing than changing the category of risk from "imposed" to "voluntary". These activities are a means to enlarge the support base to the project. When local people become supporters, they send positive, powerful and durable signals to numerous communities. Co-decisions are achievable, when participants are all acting as advocates of safety, resulting in decisions that also have symbolic meaning to the community.
- Emplacing waste in the ground poses a smaller risk than the current state of waste above ground, both now and in the future. A lack of action places the burden on coming generations. Thus, consideration needs to be given to addressing ethical bases in the safety case for a geological repository. In doing so, the safety case should acknowledge and implement the concepts of fairness, such as responsibility to present and future generations, and transparency (a full disclosure of pros and cons).

Closure of the Symposium

Closing remarks for the symposium were provided first by Hans Riotte from the NEA. The NEA applauded the supporting organisations and the hosts of the 2007 symposium: the IAEA and the EC, the French National Agency for Radioactive Waste Management and the French government. The work of the programme committee, authors, session chairs and rapporteurs was also appreciated. It was acknowledged that the symposium highlighted many important advances in numerous safety cases. Challenges still remain, further improvements can be made and the symposium underscored the value of continued international exchange on the developments, submittals and phases of the safety case.

Closing remarks by Didier Louvat from the IAEA and Wolfgang Hilden from the EC recognised the symposium as a continuation of, and an important contribution to, the important international co-operation and discussion that has been ongoing over many years regarding the safety case. The development of common concepts, principles, terminology and approaches for the safety case has advanced considerably and further harmonisation may be achieved as collaboration continues through tools such as the IAEA Joint Convention, the EC projects, and the ongoing work of the NEA IGSC.

Poster Session: Presentations on various topics related to the safety case

Authors explained their presentations and answered questions during several designated periods over the first two days of the symposium.

The poster presentations were provided by national programmes and countries representing a great variety of stages in planning and implementing geological disposal facilities. Nevertheless, there are broad similarities in basic methodology regardless of the maturity of a geological disposal programme. Emphasis was placed on site characterisation and data collection, followed by development of models of appropriate scenarios, assessment and site recommendation processes (e.g. Czech Republic, Russia, Finland and Korea). These similarities reflect that national programmes are applying a core methodology that is sound and has been internationally developed and vetted through co-operative programmes.

The following poster presentations were provided:

To what extent can natural analogues contribute to the safety case of high-level waste repositories

U. Noseck, W. Brewitz, and I. Müller-Lyda (Gesellschaft für Anlagen- und Reaktorsicherheit mbH GRS, Germany)

In this presentation, natural analogues were discussed in general terms as important lines of evidence supporting a safety case when available. Then the discussion was refined for application to the processes important to defining a repository in a rock salt dome. The most obvious analogue is the existence of other, similar rock salt domes that are very old. Near-term process analogue information was obtained from the Asse mine to support the modelling of healing in the excavation-damaged zone. For the very long term, determining the likelihood of brine intrusion, subrosion and other degrading processes was supported through the study of rock salt bodies with varying histories.

Geological repository programme in the Czech Republic – Current state of the safety case

S. Konopásková (Radioactive Waste Repository Authority, RAWRA, Czech Republic)

This presentation gave an overview of the safety case for a geological repository in the Czech Republic, which is still in its preliminary stages: pre-site-selection. A standard methodological approach is being developed for the Czech Republic repository case. The presentation described the science, technology and analogue projects under way to inform the safety case. The projects are also informed by a number of international technical cooperative projects in which Czech Radioactive Waste Repository Authority (RAWRA) has participated. This bodes well in terms of the ability of the organisation to engage in future site-specific work.

Application of archaeological analogues for repository safety case: Arguments suppor*ting the waste container lifetime*

H. Yoshikawa, K. Ueno, and M. Yui (Japan Atomic Energy Agency, JAEA, Japan)

This presentation provided an overview of arguments supporting the waste container lifetime, based on archaeological analogues. Archaeological corrosion analogues, consisting of iron artefacts that are dated up to one-thousand years old, have been found in aggressive environments near the coastline of Japan. The corrosion rates of the artefacts in these high-moisture conditions – salt spray in some instances – and aerobic conditions are less than the expected design allowance rate for the more benign repository environment. This verifies the conservatism built into the allowance rate used in safety assessments.

Application of a timeframes approach to performance assessment

M.J. Poole (United Kingdom Nirex Limited, United Kingdom)

This presentation described an approach to developing and presenting a performance assessment based on the main barriers and safety functions of a repository system and their evolution over time. The approach is designed to aid understanding and facilitate modelling on different spatial and temporal scales, in relation to the performance of the main barriers, which are defined as follows:

(1) Container.
(2) Waste Package.
(3) Chemical Barrier.
(4) Geological Barrier.

As the safety functions of the different barriers operate together and in parallel, the timeframes related to each of the barriers are regarded as 'nested' rather than sequential. For example, the geological barrier is present and providing a safety function in protecting the engineered barriers and isolating the waste, even at early times when the wastes are fully contained within the engineered barriers. The timescales for each timeframe are open-ended to reflect both the uncertainty associated with the timing of key stages in the evolution of the repository barriers and the variability associated with the duration of the safety functions.

The barriers and their safety functions are represented in the performance assessment using a hierarchy of models at different levels of detail. At the top level, these models feed into a probabilistic total system model, developed using the GoldSim software. These models will be used to calculate appropriate indicators for the performance of each of the barriers, which will be presented alongside other safety arguments in a multi-factor safety case.

Safety case for disposal facilities-confidence building and regulatory review: IAEA ASAM co-ordinated research project

M. Ben Belfadhel (Canadian Nuclear Safety Commission, CNSC, Canada), D.G. Bennett (Galson Sciences, Ltd.), P. Metcalf (International Atomic Energy Agency, IAEA, Austria), V. Nys (Association Vinçotte Nucléaire, AVN, Belgium) and W. Goldammer (Germany)

The poster relates to the IAEA project ASAM: "Application of Safety Assessment Methodologies for Near-Surface Radioactive Waste Disposal Facilities". The work of the IAEA Regulatory Review Working Group is described as providing practical tools and techniques aimed at improving the safety of radioactive waste disposal facilities. The guidance on regulatory review will provide a consistent framework and approach for a review of safety assessments and, in many respects, may also be applied to the review of safety cases. The guidance on safety assessment review can also be applied at a level of detail compatible with available resources. This is particularly important for countries with severely limited resources. The guidance on confidence building in the safety case should help in addressing the particular challenges associated with the development, upgrading and remediation of near-surface disposal facilities. To read more, the internet description of this effort was recommended: http://www-ns.iaea.org/projects/asam.htm.

^{14}C in the safety case

G. Bracke and W. Müller (Institute for Safety Technology, Germany)

This presentation addressed an approach for evaluating ^{14}C pathways in the safety case. For a repository in salt, recognition was given that ^{14}C pathways (gas and brine phase pathways) are potentially significant, and need to be assessed. The approach introduces a degree of sophistication that includes looking at release rates, at exchange of ^{14}C on solid surfaces and with dead organic carbon at depth, as well as with dissolved stable carbon and gas-phase carbon to provide a more realistic release term that considers process rates and complex processes that lead to delay of release and to isotopic dilution.

Pre-closure safety analysis for seismically initiated event sequences

M.J. Shah (Nuclear Regulatory Commission, NRC, United States of America)

This presentation demonstrated similarities between pre-closure and post-closure safety with regards to seismic analyses. A pre-closure risk assessment methodology was described with emphasis on seismic analyses. The convolution of seismic hazard and system component fragility was shown. This paper emphasised that there are similarities between aspects of pre-closure and post-closure safety assessment methodologies, and in terms of determining frequency distributions for seismic analyses.

Safety assessments of ultimate isolation of radioactive waste in deep geological formations in Russia: Objectives, problems, prospects

T.A. Gupalo, V.S. Gupalo, and V.Y. Konovalov (FSUE VNIPIpromtekhnologii, Rosatom, Russian Federation)

This presentation outlined a methodology for developing a comprehensive safety assessment. Site characterisation established initial conditions on which models are based. Using these models, key scenarios are then considered in an overall safety assessment, which provides an important input to recommendations and decision making. The need for both pre-closure safety and post-closure safety were shown to be interrelated. One unique aspect of this presentation is that the approach draws on data and experience from several underground industrial analogue sites where industrial and nuclear processes occurred over extended periods of time that included temperature and chemical transients affecting the host rock.

Fundamental processes of radionuclide migration (FUNMIG) project

Buckau, G. (FZK, Germany)

The Integrated Project "Fundamental Processes of Radionuclide Migration" (IP FUNMIG) is established within the 6^{th} framework programme of the EC. It focuses on the radionuclide-host rock interactions providing a dominant barrier between radioactive waste and the biosphere. FUNMIG tackles the scientific and social credibility of geological HLW disposal, one of the main challenges for a sustainable European energy mix where nuclear power plays an important role.

FUNMIG aims at:

- Providing tools for scientifically sound performance assessment for radionuclide migration from near-field to hydrosphere/biosphere.
- Covering the variability of different radioactive waste disposal approaches and host-rock types under investigation in Europe.
- Ensuring optimised use of resources and communication on this issue between member states with large programmes and high competence levels.
- Providing for knowledge transfer that fosters a common competence level among all European countries.
- Providing communication with national regulatory bodies responsible for the fulfillment of compliance with safety standards.
- Ensuring applicability of results for different radioactive waste disposal options and national needs.

IP FUNMIG commenced in 2005 for a duration of four years. Annual workshops are held with publication of proceedings by the respective hosting organisations. The annual workshops are open to participation by organisations not participating in the project (information: www.funmig.com).

Development of the safety case of geologic repository in Korea

Chul-Hyung Kang, Youn-Myoung Lee, Yong-Soo Hwang
(Korea Atomic Energy Research Institute, KAERI, Korea)

This presentation gave a basic overview of the Korean geological repository programme, which is in the site-selection stage. Crystalline rock is under consideration at a 500 m depth. Standard methodological approaches are being applied. A preliminary generic safety assessment has been performed that suggests the concept under consideration will meet all regulatory requirements.

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Appendix A

SESSION I

FRENCH NATIONAL PROGRAMME ON DEEP DISPOSAL SAFETY

FRENCH SAFETY RULES AND REVIEW APPROACHES REGARDING THE GEOLOGICAL DISPOSAL

P. Bodenez
Autorité de sûreté nucléaire, France

The role of the Nuclear Safety Authority

The French Nuclear Safety Authority (ASN) establishes the regulation for nuclear safety and the radiation protection matters. As a potential future nuclear installation, the development of the project of a geological disposal for high-level waste is covered by a technical rule, called the Basic Safety Rule III.2.f. This rule was established in 1991 and defines the long-term safety objectives of such disposal.

ASN authorised also the different phases of the construction of the underground laboratory sited near the Bure city, at the boarder of the Meuse and the Haute-Marne department. ANDRA applied for a licence to construct and operate this laboratory and was authorised to do so by a decree of August 1999. Since then, the drills of the shafts, and the construction of the tunnels of the laboratory were authorised after an assessment by IRSN, the Institute for Radiation Protection and Nuclear Safety.

ASN surveys, from a safety point of view, the research conducted by ANDRA. ANDRA transmits periodically the safety analysis based on the information gained through the experiments conducted in the Bure region. A dossier called "Clay 2001" was assessed by ASN, after an examination by the standing group of experts on disposal of radioactive waste. The aim of this assessment was to give recommendations for the dossier "Clay 2005" which had to be provided by ANDRA after the 15 years of research on the 3 strands set by the law of 30 December 1991, namely: storage, partitioning and transmutation, and deep geological disposal. ASN also assessed the dossier "Clay 2005" and provided an advice to the Government the 1st February 2006. This advice was published on the website of ASN.

The Basic Safety Rule RFS III.2.f on the safety of a geological disposal

The Basic Safety Rule RFS III.2.f was published in June 1991. The safety rule defines the objectives from a safety point of view during the siting, the design and the construction phase of a geological disposal. The RFS III.2.f of June 1991 was still the reference guidance used for the assessment by IRSN and the standing group of experts of the Dossier "Clay 2005".

ASN is currently updating the RFS III.2.f, in order to take into account the concept of reversibility, the experience gained in France and in other countries and the new recommendations established by international organisations like ICRP, IAEA and OECD/NEA.

The general objectives for geological disposal are the protection of human beings and the environment in the short and the long term. This protection must be assured taking into account the hazard of dispersion of radioactive contamination in different situations without depending on institutional oversight or control, which cannot be assured beyond a certain timeframe.

The concept of disposal must limit the radiological impact to a level as low as reasonably achievable taking into account technical, economic and social features. The characteristics of the site, the design of engineered barriers, and the quality of construction are the basis of the safety of the disposal. Evaluation of radiological impacts will be calculated.

In the reference situation, the dose to an individual people shall not be higher than a value of 0.25 mSv per year, for a timeframe of 10 000 years. The reference situation is a "normal evolution scenario" which includes evolution of the repository setting and deterioration of the waste package, but does not need to account for effects of low probability events. Beyond this timeframe, quantified estimations are made, and the limit of 0.25 mSv per year is kept as a reference value but not applied as a strict limit. For other situations addressing abnormal or unexpected events, the dose shall be sufficiently low compared to doses leading to deterministic effects.

The RFS III.2.f sets objectives on the concept, the packages, the engineered barriers and the geological barrier. The rule has established essential criteria on the stability of the site from the seismic and erosion point of view, and its ability to maintain radionuclides inside the host rock, through a very low permeability. Other criteria are defined as important, like the mechanical, thermal and geochemical properties of the host rock. The rule also defines a minimum depth and requires that no exceptional natural resources should be found on the site (petroleum, gas, coal...).

The safety evaluation shall consider:

- The justification of any advantageous feature on the performance of each barrier.
- The evaluation of effects of the disposal on the host rock and the verification of the acceptability of these effects.
- The evaluation of the future behaviour of the disposal and the verification that individual doses remain acceptable.

The opinion of ASN of 1 February 2006 on the research conducted since 1991 on the strands for the management of high level waste

ASN has conducted a process of evaluation through the assessment by IRSN and the standing group of experts of the safety cases provided by ANDRA, especially the safety analysis of the "dossier Clay 2005". The opinion of ASN was published in February 2006. ASN concluded that it was reasonable to search for a site in the area of 200 km² around the laboratory of Bure for a geological disposal facility. It is likely to demonstrate the safety of such disposal in the area.

ASN pointed out that some additional studies should be undertaken, like developing a strategy to investigate the 200 km² area and make progress in some particular fields. Understanding of the mechanical behaviour of the rock, particularly in relation to excavation techniques, will have to be improved by on-site experiments. The performance of the engineered seal structures will need to be confirmed and the results of the models of in-situ gas transfers and their effects on the seals will also need to be validated by on-site experiments. The soundness of the choice of disposal concepts will have to be confirmed by repository engineered structure demonstrators and safety studies, covering both the operational and post-closure phases, in particular for the ventilation systems designed to mitigate the risk of explosion due to the presence of radiolysis gases.

The operational safety arrangements, in particular the sizing of the ground supports and the ventilation system, will need to be specified. Tests in the underground laboratory will in particular be needed for recovery of waste packages from the disposal cells and for operational safety measures.

The Government took into account the recommendations of ASN when it submitted a draft law to the Parliament during the spring of 2006. The law was voted in June 2006 and finally published at the end of June 2006.

ASN prepared the decree of 23 December 2006 to authorise ANDRA to carry on the research conducted in the Bure laboratory until 31 December 2011. The law of 28 June 2006 on the sustainable management of radioactive material and waste has established a road map for the application of a license in 2015. The reversible geological disposal will be authorised as a nuclear installation by the Government, after an opinion of the CNE and ASN. The Parliament will have to precise the conditions of reversibility in a law and will authorise the closure of the disposal.

Conclusion

The law of 28 June 2006 is an important step in the development of a reversible geological disposal for HLW in France, since it is considered now a reference solution. ASN is updating its regulation on the safety of a geological disposal, which should be available in 2007. ASN will carry on the task in order to give its opinion on the safety of a project of geological disposal in the region of Bure.

Appendix B

SESSION II

THE SAFETY CASES CONCEPT AND ITS EVOLUTION

THE SAFETY CASE – CONCEPT, HISTORY AND PURPOSE

C. Pescatore
OECD Nuclear Energy Agency (NEA)

Introduction

The Nuclear Energy Agency (NEA) of the Organisation for Economic Co-operation and Development (OECD) and the International Atomic Energy Agency (IAEA) define a *safety* ***case*** as:

> "... an integration of arguments and evidence that describe, quantify and substantiate the safety, and the level of confidence in the safety, of the geological disposal facility."

The documents (NEA, 2004; IAEA & NEA, 2006) explain that the safety case draws not only on the results of quantitative modelling, but also more directly on site selection and the results of site characterisation and design studies, as well as on the research programme and management strategy by which uncertainties and open questions are to be handled.

In turn, a *safety* ***assessment*** is defined as:

> "...the process of systematically analysing the hazards associated with the facility and the ability of the site and design to provide the safety functions and meet technical requirements."

A safety case thus encompasses a safety assessment, but includes also wider qualitative arguments that together establish the basis for the level of confidence claimed in the overall safety of the disposal system.

Today, a safety case typically covers up to one million years of system evolution, and even longer in some countries. Given this timeframe, its safety assessment must not be seen as an exercise in prediction. "What must be achieved is a convincing and indirect demonstration that the proposed disposal system provides a sufficient level of safety to both current and future generations". (NEA, 1991a) It must be recognised that the ability to monitor and intervene and enforce safety measures will diminish and cannot be relied upon beyond a relatively short initial period. This poses unprecedented challenges for the safety case, to the regulator, the implementer, the public, and any interested party in general.

The present paper reviews the major milestones and consolidation stages in the development of the safety case concept since the late 1980s and the associated evolution of key elements from the perspective of over 20 years of safety-case-related work in the OECD Nuclear Energy Agency.

Major Milestones

1989 Symposium: First consolidation of the state of the art

The state of the art in safety assessment was reviewed in a 1989 Symposium (NEA, 1990) and captured in an NEA technical brochure (NEA, 1991b) on the state of the art in safety *assessment* methods and in a Collective Opinion (NEA, 1991a) on whether long-term safety can be *evaluated*. The two words *assessment* and *evaluation* give us a clue to the nature of these two publications, their respective authors, and the different challenges being addressed.

An *assessor* of safety is typically a technical specialist whose main interests lie in long-term quantitative performance analysis. According to the NEA technical brochure (NEA, 1991b) a safety assessment is based on scenarios, on models, on analysis of uncertainties, on "validation and review of all components in the assessment", and a comparison of the results is made against established criteria for acceptability. Does an assessor have the means to perform these tasks in a credible way? Whilst noting that safety assessment methods would be further developed, the NEA technical brochure (NEA, 1991b) provided a broadly positive answer to this question.

The *evaluator* of safety is closer to decision making than to the technical analysis and, while progress in safety assessment methods is acknowledged, attention is drawn in the Collective Opinion (NEA, 1991a) to three basic conditions for judging safety:

(a) the need for integrated assessments;
(b) the consideration of uncertainties in assessment results;
(c) the methods for building confidence in assessments results.

In other words, in order to form a judgement, the evaluator of safety needs more than a safety assessment, and emphasis is placed on the confidence building element: "The treatment of uncertainties in safety assessment is, however, part of a wider issue: the necessity of *building confidence* in disposal system safety. Confidence is achieved by many means. For instance, confidence is enhanced through the establishment of appropriate quality assurance and quality control procedures. . . . Expert judgment and peer reviews are also part of the confidence building process." (NEA, 1991a) Modern safety cases are meant indeed to quantify and substantiate not only safety but also "the level of confidence" in safety. In a nutshell, the evaluator needs a "safety case" that goes beyond a quantitative safety assessment.

Following on these key publications, the NEA took up the challenge and, over the next decade, worked on the important issues identified. During this time, the modern concept of safety case was formulated, which has the concept of confidence and confidence building at its core, as explained hereafter.

1994: GEOVAL '94: Exit "validation", enter "confidence building"

For over a decade the NEA had sponsored projects such as INTRAVAL and CHEMVAL, in which "VAL" meant a form of validation of computer codes in the areas of groundwater and radionuclide movement. BIOMOVS was an IAEA sponsored project to validate biospheric models for radionuclide accumulation and transfer. INTRAVAL (1988-1994) was itself a successor to two validation projects: INTRACOIN (1981-1984) and HYDROCOIN (1984-1987). The word validation was typically associated with "the genuine understanding" and mimicking of nature. [NEA, 1991b] By the early 1990s, the literature on validation and on the import of this concept was fast developing. Two major symposia dealt with model validation: GEOVAL-90 (NEA, 1991c) and GEOVAL-94 (NEA, 1995).

Whilst the various NEA and IAEA projects had provided important training and experience in model testing and the design of experiments, the utilisation of the word "validation" was proving to be confusing. Even in the same discipline the term was used in different ways and understood to mean different things. Besides, in other quarters, the use of this term was raising expectations that were not warranted. In order to understand the depth and the extent of the issues, it may be worth revisiting the GEOVAL'94 proceedings, and especially the paper by Pescatore and the verbatim transcript of the final panel session. (NEA, 1995) It became evident – mainly as a result of GEOVAL'94 – that the concept of confidence building in view of stepwise decision making emerged as the more prominent concept in lieu of validation. Later in this paper more will be said on "confidence building". It is clear, however, that the latter became an indispensable and fundamental concept for the safety case since then.

The important issues of integrated assessment and confidence building were taken up directly in two separate and eventually converging initiatives: The NEA Integrated Performance Assessment Group (IPAG) projects and the ad-hoc group on validation/confidence building. Model testing still continued in projects such as GEOTRAP (1996-2002), which resulted in important conclusions and recommendations also related to confidence building. (NEA, 2002a)

1994-2002: IPAG: Integrated assessments and their review, safety case, difficulties with multi-barrier concept

A safety assessment is much more than just collecting "good" models, assembling them, and applying them. It presupposes the integration of a number of disciplines and specialists, hence the emphasis on "integrated performance assessments". The NEA Integrated Performance Assessment Group (IPAG) was started in 1994 with the task to review the experience of performing an integrated performance assessment, and to understand what the typical elements of these assessments were. A bottom-up approach was conceived whereby an intercomparison was first made of the various national integrated performance assessment (PA) studies available up to 1996 (IPAG-1), and then of the experience of regulatory review of those studies (IPAG-2). A third study (IPAG-3) was later devoted to "confidence building" measures as seen by safety assessment experts. All these initiatives broke new ground and provided important leads.

IPAG-1 proved to be a fundamental exercise. (NEA, 1997) It identified several areas where significant advances had already taken place, and found that improvements were needed in three areas: (a) better developing traceability and transparency, (b) better interaction between PA and site characterisation; (c) treatment of spatial and temporal variability and uncertainty. It also found that there was room for "the development of a more profound general understanding". Key outcomes and innovations of IPAG-1 include that it:

- distinguished between the concepts of traceability and transparency and provided a nine-point guidance on the latter;
- gave new impetus to studies integrating the geosphere knowledge and PA (and issue also taken up in GEOTRAP);
- highlighted that PA studies have limitations that need to be recognised;
- proposed a typical skeleton for a safety assessment report, one element of which was "confidence in the key arguments".

Importantly, IPAG-1 also observed that vocabulary varied from programme to programme, and that for the purpose of understanding what was done it would be useful to introduce a hierarchy of concepts including "performance analysis", "performance assessment", "safety analysis", "safety assessment" and culminating into the concept of "*safety case*". Most studies to date were categorised as belonging to the class "safety assessment" or "safety case".

Although the term "safety case" not coined by IPAG,[1] IPAG-1 was the first attempt to provide a definition of what a "safety case" could be and how it could be structured. The term "safety case", as introduced by IPAG, was soon adopted by national programmes.

IPAG-2 was – and still is – a unique study of the actual experience of regulatory review of a safety study, including the experience of both the reviewer and the reviewee. Amongst its results [NEA, 2000], (a) it pointed to the difficulty of properly defining and utilising in practice the "multi-barrier concept", (b) it observed that qualitative information is essential in long-term safety assessments and suggested that "rather than viewing qualitative information as being inferior to quantitative information, it should be considered a different type of information that can be used for different purposes in PA". It recommended that successive studies "consider ways for both presenting and assessing qualitative arguments and information to increase their value in the decision-making process."

1996-1999: Formulation of the modern concept of safety case and its development

In 1996 the *ad hoc* group on validation/confidence building was started. Its final report constitutes a central document for the understanding of modern safety cases, which are built around the concept of attaining sufficient confidence for the decision at hand. (NEA, 1999) Sufficient confidence for a positive decision does not imply that all relevant issues have been resolved, but rather that these issues are not judged to be critical for a particular decision at hand and that there are good prospects to resolve them in future repository development stages. This should be the object of a confidence statement. The report introduced specific terminology and definitions, such as "assessment capability", "system concept", "safety case" and others that are still valid today.

The basic steps for deriving the safety case at various stages of repository development were also identified and are shown in Figure 1. They involve two major components:

1. A safety assessment, which includes:
   - the establishment of an assessment basis in which there is confidence, i.e. a strategy for the building of a safety case, selection of a site and design, and assembly of all relevant information, models and methods;
   - the application of the assessment basis in a performance assessment that explores the range of possible evolutions of the repository system and tests compliance of performance with acceptance guidelines; and
   - the evaluation of confidence in the safety indicated by the assessment and modification, if necessary, of the assessment basis;
2. The documentation of the safety assessment, a statement of confidence in the level of safety indicated by the assessment, and the confirmation of the appropriateness of the safety strategy, either in anticipation of the next stages of repository development or in response to interaction with decision makers.

The safety case should make explicit the principles adopted and methods followed in order to establish confidence. The approaches to establish confidence in the evaluation of safety should aim to ensure that the decisions taken within the incremental process of repository development are well-

1. For instance, the UK Department of the Environment (DoE) consultation document of 1994 indicated, in its paragraph 79, that "the safety case provided by the repository operator" would need to address a number of issues and demonstrate "good engineering practice, good science".

founded. Various aspects of confidence in the evaluation of safety, and their integration within a safety case, are presented in detail in the 1999 NEA report. (NEA, 1999) The key messages arising from the analysis are highlighted below.

- A safety case should make explicit the approaches that are implemented in order to establish confidence in the safety indicated by an assessment.
- The assessment basis, as defined in the report, is a key element of any safety case. In order to establish confidence in the safety indicated by an assessment, confidence in the elements of the assessment basis must be evaluated. If necessary, the elements must be modified with a view to confidence enhancement.
- Confidence evaluation and enhancement are performed iteratively in the preparation of a safety case.
- Methods exist to evaluate confidence in the level of safety indicated by an assessment, in the inevitable presence of uncertainty. In many cases, it can be determined whether safety could be compromised by specific uncertainties through a sensitivity analysis in which the consequences of such uncertainties are evaluated.
- Means exist whereby confidence in the safety indicated by an assessment can be enhanced, by ensuring the robustness of the system concept, the quality of the assessment capability, the reliability of its application in performance assessment and the adequacy of the safety strategy to deal with unresolved, safety-relevant issues.
- Observations of natural systems play an important role in the qualitative evaluation and enhancement of confidence, since such systems have evolved over extremely long time-scales.
- A statement of confidence in the overall safety indicated by the performance-assessment results is part of the safety case and should include an evaluation of the arguments that were developed, in relation to the decision to be taken.

2000-2007 Consolidation and reinforcement of the safety case concept and its elements

Since 2000, the main driver of safety case support studies has been the Integration Group for the Safety Case (IGSC) of the NEA, whose activities and publications are accessible at http://www.nea.fr/html/rwm/igsc.html. A major consolidation of the safety case "doctrine" is distilled in an IGSC brochure (NEA, 2004) which served also as a basis underlying a joint IAEA and NEA guidance on the subject. (IAEA and NEA, 2006).

Additional intervening publications that have played an important role in supporting the development of the modern concept of safety are (a) the ICRP-81 publication (ICRP, 2000) whereby the radiological exposures from geologic disposal are categorised as "potential exposures" and in which a demonstration of the application of sound engineering principles and best available techniques plays an important role for demonstrating safety. Namely, in the ICRP philosophy, radiological exposures calculations are no longer the sole basis for safety over the long term; (b) the EC SPIN Report (Becker, 2003) that supports an increased emphasis and importance for indicators other than effective dose in the long term. Table 1 shows the safety indicators reviewed in the SPIN Project.

Two additional major, recent consolidations are represented by:

- The document supporting NEA Peer Review Guidance (NEA, 2005), based on both the NEA Confidence Document of 1999 (NEA, 1999) and the experience of a score of international peer reviews. The peer review questionnaire provides a checklist of over 40 check items to review a safety assessment and the underlying assessment approach in order to enhance the quality of the safety case. (NEA, 2005)

- The IGSC "Timescales" study (NEA, 2007a), which shows the tremendous progress achieved in the last decade in safety assessment methodology and includes open issues for the safety case. Indeed, while very significant progress has been and is being made as regards technical issues, there still exist areas where additional guidance may help improve safety cases. This IGSC report constitutes the most up-to-date primer for those interested in working on the safety case. One outstanding area is how to deal with very long time scales, e.g. those that are beyond the timeframe for which meaningful calculations are possible, or, alternatively, how to introduce and justify cut-offs in time covered by such calculations. Resolution of some key outstanding issues may depend on co-operation and input from other constituencies beyond technical specialists.

Table 1: **Safety indicators and their potential uses as tested in the SPIN Project of the EC**

Indicator	Measure for system safety	Reference values		Weighting scheme		Calculable by use of PA models	Easy to understand	Added value	Biosphere pathways excluded	Dilution in aquifer excluded
		available	safety-relevant	available	safety-relevant					
Effective dose rate	+	+	+	+	+	+	+	+	-	-
Radiotoxicity concentration in biosphere water	+	+	+	+	+	+	+	+	+	-
Radiotoxicity flux from geosphere	+	+	+	+	+	+	+	+	+	+
Time-integrated radiotoxicity flux from geosphere	+	+	-	+	+	+	-		+	+
Radiotoxicity outside geosphere	+	+	-	+	+	+	-		+	+
Relative activity concentration in biosphere water	+	-		-		+	+		+	-
Relative activity flux from geosphere	+	-		-		+	+		+	+

A major, recent initiative of the NEA Regulators' Forum on the Long-term Safety Criteria (LTSC) – partially covered in the proceedings of this Symposium (see paper by R. Ferch) – provides important complementary information to the "Timescales" study (NEA, 2007a) and concluding by identifying many of the same issues. Overall, three main conclusions arise from the LTSC work:

- There are important variations in numerical criteria for long-term disposal safety in NEA countries. The quantitative differences should have no significant consequences in terms of radiological impact, however, as the criteria represent only a small fraction of natural levels of radiation exposure. Besides, it should be borne in mind that the calculated doses and risks that are measured against these criteria are only indicators of performance. Protection requirements associated with complementary measures such as optimisation and the application of "best available techniques not entailing excessive costs" are equally important.

- There are variations in the bases for criteria and the ways they are used to demonstrate the achievement of fundamental safety goals. This variation is grounded in societal differences and makes it difficult to compare directly different national regulatory approaches.
- Developing a common understanding of obligations to future generations and of how to implement these obligations in regulatory criteria for long-lived radioactive waste would make comparisons of regulatory approaches within national and international contexts, including at IAEA Joint Convention review meetings, more meaningful and useful.

Finally, one new initiative that should provide additional information on the production of the safety case is the ongoing INTESC (INTernational Experience in developing Safety Cases) project of the IGSC, which has adopted a similar working method and approach as was implemented for IPAG. The reader is advised to keep abreast of its developments.

Major Developments since the late 1980s

Safety assessments have evolved profoundly since the early 1990s. Specifically,

- There is now a strong trend to build and show detailed system understanding, with programmes having a separate set of documents, e.g. to gather and demonstrate the accumulated scientific knowledge base.
- The FEP (**F**eatures, **E**vents, and **P**rocesses) data base of the NEA has been for many years now the reference data base from which to draw scenarios for safety assessments. Following the strong trend to build and use detailed system understanding, a number of recent safety assessments do not start any longer from an analysis of externally generated FEPs. Rather "scenarios" are generated internally to the project based on the scientific and technical knowledge base that the project has accumulated. This knowledge base may differ in detail in time and space and is accordingly differently treated. In these cases, the NEA FEP data base is typically used post-hoc for completeness checking.
- An assessment of the performance of components of the barrier system in terms of required safety functions is becoming more widespread, alongside the earlier approach of assessing nuclide transport through the entire (multi-barrier) disposal system. Performance indicators can be associated with different safety functions. Qualitative analysis can also be performed.
- "What-if" scenarios are introduced in several programmes, which allow production of an upper bound to what is realistically possible. That is, they address situations or events that are not necessarily physically impossible, but which lie outside the range of possibilities reasonably expected to occur according to the available scientific understanding. As such, they can be useful tools for testing the resilience and integrity of the repository concept.
- Traceability and transparency have greatly improved through reproducible documentation and documentation structure and templates.
- There is recognition of the value of using multiple lines of reasoning to strengthen any argument that is made, to the extent practicable.
- There typically is acknowledgement of developments internationally.
- Quality systems are typically implemented, including managerial approaches to reduce uncertainties such as bias audits performed by different teams within the same organisation, etc.

- There is agreement on the central role of the concept of confidence building, although a certain difficulty is still to be seen in its application to its fullest extent. (Pescatore, 2005)
- There is agreement that safety cases will be developed in steps along with the stepwise development of the repository and that these iterative safety cases will form the basis of discussions amongst all interested parties before each decision is taken. To this effect, confidence building measures such as the ones described in the NEA documents already discussed (NEA, 2002a; 2002b; 2005) – plus communication and the development of shared confidence – will be important to obtain safety cases that will allow the building, operating and closure of a repository. These interactions and the idea of stepwise development are captured in Figure 2.

Conclusions

Major developments have taken place over the last two decades both for safety assessments and safety cases. The concept of a safety case was introduced as a way to allow safety evaluators and decision-makers to make a judgement about safety.

The concepts of confidence and confidence building are central to both the safety case and to decision making, and underlie much of the work being performed to date. New tools and new approaches have been introduced and there has been important evolution of concepts.

Overall, the safety cases currently being produced resemble more and more the desiderata of the NEA confidence document (NEA, 1999) and of the safety case brochure (NEA, 2004) and, therefore, of safety evaluators and decision-makers.

The NEA has been at the core of developments in the past 20 years. The IGSC group continues this tradition of co-operation with other international *fora*, such as groups within the IAEA and within the NEA itself, such as the NEA Regulators' Forum and the Forum on Stakeholder Confidence.

Throughout the past 20 years much effort has been applied to integrating the viewpoints of a multitude of technical disciplines and regulatory input towards creating better safety cases. More work on regulatory and societal expectations should be foreseen in the years to come, while adding more and more the experience from practical construction and operation of the underground facilities.

Special acknowledgements

The safety case is a relatively new discipline, yet it must mourn already some of its founders. Special acknowledgements are due to contributions of two colleagues who are no longer with us: Timo Vieno, who, amongst other things, directed the IPAG-1 exercise, and Richard Storck, who, amongst other things, directed the SPIN Project. Both colleagues were members of the NEA Performance Assessment Advisory Group (PAAG) that spearheaded much of the progress in the safety case area in the 1990s.

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THE EVOLUTION OF THOUGHTS FROM ICRP 46 CONCEPT OF POTENTIAL EXPOSURE

A. Sugier
ICRP Chairperson of Committee 4

Introduction

Since issuing its latest basic recommendations in 1991 as ICRP *Publication 60* (ICRP, 1991b), the Commission has reviewed these recommendations regularly and, from time to time, has issued supplementary reports in the *Annals of the ICRP*. The extent of these supplementary reports has indicated a need for consolidation and rationalisation. New scientific data have also been published since *Publication 60,* and while the biological and physical assumptions and concepts remain robust, some updating is required.

In addition, there have been societal developments in that more emphasis is now given on the protection of individuals and stakeholder involvement in the management of radiological risk. Finally, it has also become apparent that the radiological protection of non-human species should receive more emphasis than in the past.

It is against this background that the Commission has now decided to adopt a revised set of Recommendations while at the same time maintaining stability with the previous recommendations. Following several years of an open and worldwide discussion process, mainly through web consultation, ICRP intends to publish its new recommendations in 2007.

In the context of AEN/NEA seminar on "safety case for the deep disposal of radioactive waste", it appeared necessary first of all to examine the above mentioned evolution of ICRP system, as well as to recall the main ICRP publications on potential exposure and waste disposal and finally to focus on the main recommendations on solid waste disposal which are still valid (Publication 81).

Evolution of ICRP system of radiological protection

The objective of the revision of ICRP general recommendations was mainly to take account of new biological and physical information and of trends in the setting of radiations safety standards as well as to improve and streamline the presentation of the recommendations while maintaining as much stability in the recommendations.

As far as the system of protection is concerned, the general orientation adopted in the new recommendations is to focus on ICRP main message. Based on the linear non threshold model what has to be understood is that there is a continuum of risk whatever the level of exposure. Although there is not a boarder line between safe and unsafe level of exposure, the risk each of us is ready to accept is dependant on the context of the exposure. Thus an international guidance is needed to help decision makers in selecting such levels of protection. These should be source-related as what is operational is carried out at the source.

Because of the variety of radiation exposure situations and of the need to achieve a consistency across a wide range of applications, the Commission has established in the past a formal system of radiological protection aimed at encouraging a structured approach to protection. The system has to deal with a large number of sources of exposure, some already being in place, and others that may be introduced deliberately as a matter of choice by society or as a result from emergencies. These sources are linked by a network of events and situations to individuals and groups of individuals comprising the present and future populations of the world. The system of protection has been developed to allow this complex network to be treated by a logical structure.

The Commission has previously distinguished between practices that add doses and interventions that reduce doses (ICRP, 1991b). The principles of protection have been formulated somewhat differently in the two cases. Many have seen the distinction between them as artificial. Therefore, the Commission now uses a situation based approach to characterise the possible situations where radiation exposure may occur as planned, emergency, and existing exposure situations; and applies one set of fundamental source-related principles of protection for all of these situations.

In these new recommendations, the Commission emphasises the optimisation of protection regardless of the type of source, exposure situation or exposed individual. Source-related restrictions on doses or risks are applied during the optimisation of protection. In principle, protective options that imply doses above the level of such restrictions should be rejected. However in specific set of circumstances, particularly in an emergency or an existing exposure situation, it could be the case that no viable protective option can immediately satisfy the level of protective action selected from generic considerations. Thus a progressive approach should be adopted to comply with the above mentioned restrictions levels. *These restrictions are called constraints in case of planned exposure and reference levels in case of emergency and existing situations*.

In its new recommendations, ICRP has clarified how the principles of radiological protection (justification, optimisation and limitation) apply to sources and to individuals. The following wording has been adopted:

Source related principles (applying whatever the situation):

- **The principle of justification:** Any decision that alters the radiation exposure situation should do more good than harm.
- **The principle of optimisation of protection:** the likelihood of incurring exposures, the number of people exposed and the magnitude of their individual doses, should all be kept as low as reasonably achievable, taking into account economic and societal factors**.**

Individual related principle (applying only in planned exposure situation):

- **The principle of application of dose limits in planned situations:** The total dose to any individual from all planned exposure situations other than medical exposure should not exceed the appropriate limits specified by the Commission.

The chosen value for a constraint or a reference level will depend upon the prevailing circumstances of the exposure under consideration. It must also be realised that neither of them represent a demarcation between “safe” and “dangerous” or reflect a step change in the associated health risk for individuals Taking into account the quantified values of dose restrictions in previous ICRP publications, the Commission has proposed a framework for source-related constraints and reference levels from 0,01 mSv/year to 100 mSv/year (acute or annual dose). According to the characteristic of the situation and to the associated requirements (individual benefit or societal one, information, training, etc.), a more or less stringent restriction can be selected.

There are no major modifications in the concept of potential exposure which is part of planned exposure situations.

Main ICRP publications on potential exposure

The term “potential exposures” refers to situation where there is a potential for exposure but no certainty that it will occur. Such exposures may result following deviations from planned operating procedures, accidents including the loss of control of radiation sources and malevolent events. Protection from potential for exposures should be considered specifically but not only at the planning stage and may lead to actions both to reduce the probability of the events occurring, and limit and reduce the exposure (mitigation) if any event were to occur (ICRP, 1991; 1997).

Potential exposure broadly covers three types of events:

- Events where the potential exposures would primarily affect individuals who are also subject to planned exposures. The number of individuals is usually small, and the detriment involved is the health risk to the directly exposed persons. The processes by which such exposures occur are relatively simple, e.g., the potential unsafe entry into an irradiation room. The Commission has given specific guidance for the protection from potential exposures in *Publication 76* (ICRP, 1997). This guidance remains valid.
- Events where the potential exposures could affect larger number of people and not only involve health risks but also other detriments, such as contaminated land and the need to control food consumption. The mechanisms involved are complicated and an example is the potential for a major accident in a nuclear reactor or the malicious use of radioactive material. The Commission has provided a conceptual framework for the protection from such type of events in *Publication 64* (ICRP, 1993). This framework remains valid. In *Publication 96* (2005a), the Commission provides some additional advice concerning radiological protection after events involving malicious intent.
- Events in which the potential exposures could occur far in the future, and the doses be delivered over long time periods, e.g., in the case of solid waste disposal in deep repositories. Considerable uncertainties surround exposures taking place far in the future. Thus dose estimates should not be regarded as measures of health detriment beyond times of around several hundreds of years into the future. Rather they represent indicators of the protection afforded by the disposal system. The Commission has given specific guidance for the disposal of long-lived solid radioactive waste in *Publication 81* (ICRP, 1998c). This guidance remains valid.

The evaluation of potential exposures, for the purpose of planning or judging protection measures, is usually based on: a) the construction of scenarios which are intended typically to represent the sequence of events leading to the exposures; b) the assessment of probabilities of each of these sequences; c) the assessment of the resulting dose; d) the evaluation of detriment associated with that dose; e) comparison of the results with some criterion of acceptability; and f) optimisation of protection which may require several reiterations of the previous steps.

The principles of scenario construction and analysis are well known and are often used in engineering. Their application was discussed in *Publication 76* (ICRP, 1997). Decisions on the acceptability of potential exposures should take account of both the probability of occurrence of the exposure and its magnitude. In some circumstances, decisions can be made by separate consideration of these two factors. In other circumstances, it is useful to consider the individual probability of radiation-related death, rather than the effective dose (ICRP, 1997). For this purpose, the probability is defined as the product of the probability of incurring the dose in a year and the lifetime probability of radiation-related death from the dose conditional on the dose being incurred. The resulting probability can then be compared with a risk constraint. Both of these approaches are discussed in the Commission's recommendations for the disposal of long-lived solid radioactive waste in *Publication 81* (ICRP, 1998c).

Risk constraints, like dose constraints, are source-related and in principle should equate to a similar health risk to that implied by the corresponding dose constraints for the same source. However, there can be large uncertainties in estimations of the probability of an unsafe situation and the resulting dose. Thus, it will often be sufficient, at least for regulatory purposes, to use a generic value for a risk constraint based on generalisations about normal occupational exposures, rather than a more specific study of the particular operation. Where the Commission's system of dose limitation has been applied and protection is optimised, annual occupational effective doses to an average individual may be as high as about 5 mSv in certain selected types of operation (UNSCEAR, 2000). For potential exposures of workers, the Commission therefore continues to recommend a generic risk constraint of $2\ 10^{-4}$ per year which is similar to the probability of fatal cancer associated with an average occupational annual dose of 5 mSv (ICRP, 1997). For potential exposures of the public, the Commission continues to recommend a risk constraint of $1\ 10^{-5}$ per year, corresponding to the probability of fatal cancer associated with the generic dose constraint of 0.3 mSv applied e.g. in the case of disposal of long-lived radioactive waste (ICRP, 1998c).

The use of probability assessment is limited by the extent that unlikely events can be forecast. In circumstances where accidents can occur as a result of a wide spectrum of initiating events, caution should be exercised over any estimate of overall probabilities because of the serious uncertainty of predicting the existence of all the unlikely initiating events. In many circumstances, more information can be obtained for decision making purposes by considering the probability of occurrence and the resultant doses, separately.

Specific case of long level solid radioactive waste

Publications on waste

ICRP has issued three specific publications on waste disposal:

- **ICRP 46: Radiation Protection Principles for the Disposal of Solid Radioactive Waste**, published in 1985. It is in this publication that ICRP, for the first time, makes a distinction between the exposures resulting from "normal developments" and those resulting from "probabilistic events" (the word "potential exposure" was not yet used) to implement the process of optimisation, so as to select these options for a reposition. Although the publication ICRP 46 is considered as still valid, its approach is now considered as too theoretical, taking into account the uncertainties in the assessments both of the doses and the probabilities in the case of long term disposal. ICRP 46 emphasised too strictly the requirement of compliance between calculated risks with risk limits and risk upper bounds – now called constraints. Moreover it was recommended that the final step of such an optimisation process should be "to check, whether the option of protection for the repositions meets the ultimate requirement that the total risk from all scenarios, resulting from their combination, should comply with the risk upper bound condition".

- **ICRP 77: Radiological Protection Policy for the disposal of radioactive Waste**, published in 1997. It is a more general framework emphasising the different strategies for waste disposal, the problems raised by an excessive aggregation of collective dose over time, the limitation of the relationship between dose and detriments for time periods of more than hundreds years. For the first time, ICRP recommended a value for the dose constraint for public exposure (0.3 mSv/year $^{-1}$).
- **ICRP 81: Supplement, update and clarify ICRP 46 recommendations** (published in 1999). Although it was issued ten years ago, this publication is completely in line with the orientations of the new recommendations, as well as with the foundation documents on optimisation and on the representative individual.

Main orientations in ICRP 81

Constrained optimisation is the central approach to evaluating the radiological acceptability of a waste disposal system; dose or risk constraints are used rather than dose or risk limits. By this transition, from limits towards individual restrictions to optimisation, the needs of practical application of the radiological protection system to the disposal of long-lived solid waste disposal are met: determination of acceptability *now* for exposures that may occur in the distant *future*. Optimisation should be applied in an iterative manner during the disposal system development process and should particularly cover both site selection and repository design.

Two broad categories of exposure situations should be considered: natural processes and human intrusion. Both are considered as giving rise to potential exposures. The latter only refers to intrusion that is inadvertent. The radiological implications of deliberate intrusion into a repository are the responsibility of the intruder. Assessed doses or risks arising from natural processes should be compared with a dose constraint of 0.3 mSv per year or its risk equivalent of around 10^{-5} per year. With regard to human intrusion, the consequences from one or more plausible stylised scenarios should be considered in order to evaluate the resilience of the repository to such events.

The Commission 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 at the repository development stage to reduce the probability of human intrusion or to limit its consequences. In this respect, the Commission has previously advised that an existing annual dose of around 10 mSv per year may be used as a generic reference level below which intervention is not likely to be justifiable (ICRP 82 on Protection of the Public in Situations of Prolonged Radiation Exposure, extends exposure). Conversely, an existing annual dose of around 100 mSv per year may be used as a generic reference level above which intervention should be considered almost always justifiable. Similar considerations apply in situations where the thresholds for deterministic effects in relevant organs are exceeded.

Compliance with the constraints can be assessed by utilising either an aggregated risk-oriented approach, with a risk constraint, or a disaggregated dose/probability approach, with a dose constraint, or a combination of both. A similar level of protection can be achieved by any of these approaches; however, more information may be obtained for decision-making purposes from the disaggregated approach.

Demonstration of compliance with the radiological criteria is not as simple as a straightforward comparison of calculated dose or risk with the constraints, but requires certain latitude of judgement. Neither should estimated transgression of a constraint necessarily oblige rejection, nor should numerical compliance alone compel acceptance of a waste disposal system. The dose or risk constraints should increasingly be considered as reference values for the time periods farther into the future, and additional arguments should be duly recognised when judging compliance.

Application of technical and managerial principles during the disposal system development process will enhance confidence in the safety provided by the disposal system. These principles should be based on those elaborated by the Commission for application in potential exposure situations.

In the Commission's view, provided that reasonable measures have been taken both to satisfy the constraint for natural processes and to reduce the probability or the 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.

Conclusion

Application of potential exposure concept to exposures that could occur far in the future is a particular difficult issue because of the level of uncertainties associated with the assessment of the probabilities of the events leading to the exposures as well as with the evaluation of the doses to the populations.

The general orientations of the new ICRP recommendations aimed at a less complex system focusing on the main message: whatever the dose there is a risk, and proposing a simple scale of source related constraints or reference levels and focusing on a judgmental approach of optimisation, is better suited to such an issue.

ICRP 46, although too theoretical in its approach, was a kind of pioneer by applying optimisation to probabilistic events and setting upper levels to the process. In the same way, ICRP 81, in recommending a more judgmental approach focusing on a disaggregated and judgmental approach of optimisation based on dose and probability estimations to be compared with dose or risk constraints is in line with the new recommendations (RP07).

THE EVOLUTION OF SAFETY STANDARDS (WS-R-4) AND REQUIREMENTS IN RESPECT OF THE SAFETY CASE

P. Metcalf
International Atomic Energy Agency (IAEA)

Abstract

In terms of its statute the IAEA is mandated to develop safety standards for nuclear, radiation, radioactive waste and transport safety, and to provide for their use and application. At the 2000 Cordoba Conference on the safety of radioactive waste management it was concluded that in view of foreseen developments in the area that work should commence on the development of safety standards for geological disposal. Work commenced on a Safety Requirements standard in 2001 and was concluded in 2005 with the adoption of the standard by the IAEA Board of Governors and the Steering Committee of the OECD/NEA, co-sponsors of the standard. The standard sets down protection criteria together with a comprehensive set of discrete requirements for the planning, development, operation and closure of geological disposal facilities. Specific requirements are included regarding the preparation of the safety case and safety assessment, the scope of the safety case and safety assessment and concerning documentation of the safety case and safety assessment. The paper elaborates the requirements, provides some background to their development and discusses activities of the IAEA related to their use and application.

The IAEA safety standards for nuclear, radiation, radioactive waste and transport safety

In terms of its statute [1], the IAEA is mandated *to establish or adopt, in consultation and, where appropriate, in collaboration with the competent organs of the United Nations and with the specialised agencies concerned, standards of safety for protection of health and minimisation of danger to life and property (including such standards for labour conditions), and to provide for the application of these standards to its own operation as well as to the operations making use of materials, services, equipment, facilities, and information made available by the Agency or at its request or under its control or supervision; and to provide for the application of these standards, at the request of the parties, to operations under any bilateral or multilateral arrangements, or, at the request of a State, to any of that State's activities in the field of atomic energy.*

Since establishment of the IAEA in 1957 its process of safety standards development has evolved and matured into an ongoing process involving development of new standard and continual review and updating of existing standards covering all aspects of nuclear, radiation, radioactive waste and transport safety. The process involves all 143 Member States of the IAEA and is controlled and directed by them. The standards are consensual and approved by the Board of Governors of the IAEA. Many of the safety standards are developed in co-operation with other relevant UN bodies and international and regional organisations such as the OECD/NEAand the EC, and cosponsored by them. The standards are presented in a suite of documents addressing particular facilities and activities such as nuclear power reactors and radioactive waste disposal facilities or thematic areas such as legal and governmental infrastructure and radiation protection. The safety standards documents are also structured in a hierarchy; Safety Fundamentals, Safety Requirements and Safety Guides. A single Safety Fundamentals document [2] provides the basic principles underlying all the nuclear, radiation, radioactive waste and transport safety standards. A number of Safety Requirements documents provide the technical, legal and procedural imperatives necessary to fulfill the principles; these are supported with Safety Guides providing recommendations and guidance to meet the requirements for various facilities and activities. The recommendations and guidance represents prevailing best practice, which necessarily evolves with time; one of the reasons to review and update the standards on an ongoing basis. Apart from the need to update standards to reflect current best practice and to improve their presentation and utility, their development is influenced by developments within the nuclear industry and the use of radiation related technologies. Considerable attention was given to standards for nuclear safety following the Chernobyl accident, which also led to the establishment of the Convention on Nuclear Safety [3], which in turn led to the Joint Convention on the Safety of Spent Fuel Management and the Safety of Radioactive Waste Management [4]. The development and entry into force of the Joint Convention focused increasing attention on the radioactive waste safety standards.

The safety standards are used for a number of purposes. They are often used by Member States in the development and updating of legislation and regulatory guidance, they are used to assist in the development of facilities and particularly in licensing processes, either in setting regulatory requirements or as a basis for regulatory review of safety submissions. They are also used extensively at an international level as a basis for international peer review, in the international safety conventions and, particularly in the case of transportation, as a basis for related international legal instruments. The standards are also used extensively as training tools in nuclear, radiation, radioactive waste and transport safety and to focus and direct international inter-comparison and safety harmonisation projects

Development of WS-R-4, the Safety Requirements for Geological Disposal

As indicated above, the development and coming into force of the Joint Convention focused international attention on related safety standards, one area in particular being identified by the 2000 Cordoba Conference; *The Safety of Radioactive Waste Management*, was geological disposal. The conference recognised the developments taking place around the world in this area and amongst the actions arising from the conference was a request from Member States of the IAEA to develop safety standards for geological disposal. The process commenced in 2001 in co-operation with NEA/OECD and a Safety Requirements standard on geological disposal was approved in 2005 [5].

The development process was commenced with an international workshop to identify those areas deemed necessary for inclusion in the standard. A series of international technical meetings took place to advise on progress with the standard and the Waste Safety Standards Committee (WASSC) undertook its oversight, review and approval role. Important decisions taken were; to set down a clear safety objective and health and safety criteria, to structure the document to cover requirements to be fulfilled as a basis for developing a geological disposal facility, those to be fulfilled in its development i.e. the siting, design, excavation/construction, operation and closure of facilities and those for assuring safety. It was also agreed that the standard should cover safety during the operational period as well as post closure safety

The standard developed addresses the need for a safety case and supporting safety assessment, indicating that their development for review by the regulator and other interested parties are central to the development, operation and closure of a geological disposal facility. It is indicated that the safety case substantiates the safety, and contributes to confidence in the safety of the facility and is an essential input to all the important decisions concerning the facility. It is indicated to include the output of safety assessments together with additional information, including supporting evidence and reasoning on the robustness and reliability of the facility, its design, the design logic, and the quality of safety assessments and underlying assumptions. It points out that the safety case may also include more general arguments relating to the need for the disposal of radioactive waste, and information to put the results of the safety assessments into perspective and that unresolved issues at any step in the development, operation and closure of the facility will be acknowledged in the safety case and guidance for work to resolve these issues provided at a later stage.

The standard describes safety assessment as the process of systematically analysing the hazards associated with the facility and the ability of the site and the design of the facility to provide for the safety functions and to meet technical requirements. It includes quantification of the overall level of performance, analysis of the associated uncertainties and comparison with the relevant design requirements and safety standards. Assessments are identified to be site specific, since geological systems, in contrast to engineered systems, cannot be standardised and as site investigations progress, safety assessments become increasingly refined, and at the end of a site investigation sufficient data will be available for a complete assessment. Safety assessments also identify any significant deficiencies in scientific understanding, data or analysis that might affect the results presented. Depending on the stage of development, it is indicated that safety assessments may be used to aid in focusing research, and their results may be used to assess compliance with the various safety objectives and criteria.

Three discrete requirements statements are included in the standard:

- **Requirements concerning preparation of the safety case and safety assessment**

  "A safety case and supporting safety assessment shall be prepared and updated by the operator, as necessary, at each step in the development, operation and closure of the geological disposal facility. The safety case and safety assessment shall be sufficiently detailed and comprehensive to provide the necessary technical input for informing the regulatory and other decisions necessary at each step."

- **Requirements on the scope of the safety case and safety assessment**

  "The safety case for a geological disposal facility shall describe all the safety relevant aspects of the site, the design of the facility, and the managerial and regulatory controls. The safety case and its supporting assessments shall illustrate the level of protection provided and shall provide assurance that safety requirements will be met."

- **Requirements concerning documentation of the safety case and safety assessments**

  "The safety case and its supporting safety assessments shall be documented to a level of detail and quality sufficient to support decisions to be made at each step and to allow for their independent review."

By way of explanation, the standard indicates that a site specific safety case will be prepared early in the development of a geological disposal facility to provide a basis for licensing decisions, and to guide activities in research and development, siting and design. The safety case is progressively developed and elaborated as the project proceeds, and is presented at each key step in the development of the geological disposal facility. The regulatory body may mandate an update of or revision to the safety case before given steps, or such an update or revision may be necessary to gain political or public support for taking the next step in the development and operation of the geological disposal facility. The formality and level of technical detail of the safety case depend on the stage of development of the project, the decision in hand, the audience to which it is addressed and specific national requirements.

It is also pointed out that a safety assessment in support of the safety case will be performed and updated throughout the development and operation of the geological disposal facility and as more refined site data become available. Safety assessment are to provide input to continuing decision making by the operator, such as decision making relating to subjects for research, development of the capability for assessment, the allocation of resources, and the development of waste acceptance criteria. Safety assessments are also to identify key processes relevant to safety, contribute to the development of an understanding of the performance of geological disposal facilities, and support judgements with regard to alternative management options as an element of optimising protection and safety. Such an understanding is indicated to provide the basis for the safety arguments presented in the safety case. It is incumbent on the operator to decide on the timing and level of detail of the safety assessment, in consultation with and subject to the approval of the regulatory body.

As indicated the standard covers both operation and post closure safety and as such calls for the safety case to address both. All aspects of operation relevant to radiation safety are to be considered, including underground development work, waste emplacement, and backfilling, sealing and closing operations. Consideration is to be given to both occupational exposure and public exposure resulting from normal operations, including operational occurrences anticipated to occur over the operating lifetime of the geological disposal facility. Accidents of a lesser frequency but with significant radiological consequences, that is, accidents that could give rise to radiation doses over the short term in excess of the annual dose limits specified, are to be considered with regard to both their likelihood of occurrence and the magnitude of possible radiation doses. The adequacy of the design and operational features are also to be evaluated.

With regard to post-closure safety, the expected range of possible developments affecting the geological disposal system and the low probability events that might affect its performance are to be considered in the safety case and in the supporting assessment, by:

- Presenting evidence that the geological disposal system, its possible evolutions and relevant events that might affect it are sufficiently well understood.
- Demonstrating the feasibility of implementing the design.
- Providing convincing estimates of the performance of the geological disposal system and a reasonable level of assurance that all the relevant safety requirements will be complied with and that radiation protection has been optimised.
- Identifying and presenting an analysis of the associated uncertainties.

The safety case may include the presentation of multiple lines of reasoning based, for example, on studies of natural analogues and palaeo-hydro-geological studies, the quality of the site, the properties of the host rock, engineering considerations and operational procedures, and institutional assurances.

Regarding safety assessments, the standard requires that they analyse the performance of the geological disposal system under the expected and less likely evolutions and events, which can be outside the designed performance range of the geological disposal facility. A judgement of what is to be considered as the expected evolution and less likely evolution is expected to be discussed by the regulatory body and the operator. Sensitivity analyses and uncertainty analyses are to be undertaken to obtain an understanding of the performance of the geological disposal system and its components under a range of evolutions and events. The consequences of unexpected events and processes may be explored to test the robustness of the geological disposal system. In particular, the resilience of the geological disposal system is to be assessed. Quantitative analyses are to be undertaken, at least over the time period for which regulatory compliance is required, nevertheless it is emphasised that the results from detailed models of safety assessments are likely to be more uncertain for time periods in the far future. For such timeframes, the standard points out that arguments may be needed to illustrate safety, on the basis, for example, of complementary safety indicators, such as concentrations and fluxes of naturally occurring radionuclides and bounding analyses.

The management systems established to provide an assurance of quality in the design and operational features are to be addressed in the safety case.

The necessary scope and structure of the documentation setting out the safety case and its supporting safety assessments will depend on the step reached in the project and on the national requirements. It will include consideration of the needs of different interested parties for information. Important considerations are justification, traceability and clarity. In this regard justification concerns explaining the basis for the choices that have been made and the arguments for and against the decisions, especially those decisions concerning the main safety arguments. Traceability refers to the ability of an independent qualified person to follow what has been done. Good traceability is essential to enable technical and regulatory review. Justification and traceability both require a well documented record of the decisions made and the assumptions made in the development and operation of a geological disposal facility, and of the models and data used in arriving at a particular set of results for the safety assessments. Clarity refers to good structure and presentation at an appropriate level of detail so as to facilitate an understanding of the safety arguments. It requires that material be presented in the documents in such a way that interested parties can gain a good understanding of the safety arguments and their basis. Different styles and levels of documentation may be required in order to provide material that is useful to different parties.

Application and use of the standards

At the present time a Safety Guide is being developed to set down recommendations and guidance representing best practice for meeting the requirements set out in WS-R-4. The standard is being used by the IAEA as a basis for providing assistance to countries contemplating the development of geological disposal facilities. Experience in this regard has identified some difficulties with the concept of a "safety case", nevertheless, the comprehensive coverage of safety requirements within WS-R-4 provides a good basis for presenting the various elements that have to be addressed in the safety case.

The standard is also finding use in the work underway in Europe exploring the possibility to establish harmonised regulatory expectations of the safety case for geological disposal [6]. The work is related to that of the Western European Nuclear Regulatory Authorities (WENRA), where safety reference levels have been developed for nuclear safety and are under development for radioactive waste and spent fuel storage using the IAEA safety standards as the point of reference [7].

A number of international inter-comparison and harmonisation projects are undertaken by the IAEA which have addressed safety assessment methodology for predisposal management of radioactive waste, decommissioning, near surface disposal and environmental transport pathway modelling. In view of the increasing interest in geological disposal a similar project has been developed for geological disposal and will commence during the course of 2007. It is envisaged the project will provide a good forum for exchange of experience and inter-comparison between countries more recently contemplating geological disposal and those with more experience.

It is also envisaged that WS-R-4 will increasingly provide a point of reference for international peer review and reports to the Joint Convention relating to geological disposal facilities and activities in countries party to the convention.

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A COMPARISON OF NATIONAL DOSE AND RISK CRITERIA FOR DEEP DISPOSAL OF LONG-LIVED RADIOACTIVE WASTE

R. Ferch
Consultant,[1] Canada

Abstract

The Regulators' Forum of the NEA Radioactive Waste Management Committee (RWMC) undertook, as one of its first actions after it was formed in 1999, a comparative study of the regulation of proposed repositories for long-lived high-level waste in member countries. One particular aspect of this study was a comparison of the dose and risk criteria proposed or used in member countries, and of whether and how these criteria depended on the time scale.

The first phase of this study revealed that while, broadly speaking, the criteria used in most countries were generally similar, when these criteria were expressed in comparable units the entire range spanned was fairly broad (about two orders of magnitude). The differences are particularly evident when the criteria at long time scales (beyond 10^3-10^4 years) are compared.

This paper presents the results of this study in tabular form, together with a discussion of some of the possible underlying reasons for the differences between national criteria and of their significance. Work is ongoing to investigate these reasons with the aim of developing a common understanding of the fundamental objectives of the regulation of disposal in order to facilitate discussion of comparative approaches.

Introduction

An important component of the overall safety case preparation is the setting of the criteria against which safety assessment outputs will be evaluated by the regulatory body or other decision makers. A comparison of regulatory practices and regulatory criteria is therefore one aspect of any comparative discussion on safety cases.

When the Regulators Forum of the RWMC was formed in 1999, one of its first tasks was to review the arrangements in member countries for regulation of radioactive waste management. This work resulted in a comparative study of regulatory structures in member countries [1]. One part of the work leading to this comparative study was a review of the long-term radiological protection criteria for disposal of long-lived waste, and an examination of their consistency amongst countries.

After this initial comparison, which revealed a broad range of differing criteria and practices, an initiative on Long-Term Safety Criteria (LTSC) was undertaken, and a group was formed to examine this question in more detail [2]. This group includes representation from the Forum on Stakeholder

1. Work performed on behalf of the NEA Secretariat (C. Pescatore).

Confidence and the IGSC as well as from the Regulators' Forum. The objective of this ongoing initiative is neither to set nor to judge existing standards, but rather to study the criteria used by various member countries and to provide a forum for discussion. The scope of the comparison study in the present paper is limited to radiological protection criteria (i.e. calculated dose and risk constraints and targets) used to evaluate proposals for geological disposal of high-level radioactive waste.

The first step in the LTSC review was a questionnaire which was sent to members of the RWMC's Regulators Forum in 2004. At first there seemed to be a "good degree of agreement" that there is "broad consistency" among the target levels for protecting future generations. The key word, however, is "broad". In the table of national criteria, the dose constraints listed varied from 0.1 mSv/a to 0.3 mSv/a, i.e. by a factor of three. While not all of the responses mentioned risk constraints, those that did were split almost equally between two values of risk: 10^{-5} per year and 10^{-6} per year. Applying a nominal radiological risk conversion factor in the range of $5x10^{-2}$ to $7x10^{-2}$ Sv^{-1}, we find that dose constraints of 0.1 to 0.3 mSv/a for high-probability scenarios correspond to annual radiological risks between about $5x10^{-6}$ and $2x10^{-5}$, so the entire range in the table (from a 10^{-6} per year risk constraint to an 0.3 mSv/a dose constraint) appeared to cover a factor of roughly 20 between the highest and lowest values. Since 2004, this range has been broadened considerably by the proposed 3.5 mSv/a criterion in the US for the Yucca Mountain project for times greater than 10^4 years.

In order to interpret this range, it proves useful to consider a number of topics, among them: terminology and interpretation; the bases for selection of criteria; and how conformity with the criterion is assessed. Having better understood how to interpret the information, we can then proceed to consider whether the differences are significant in terms of radiological protection.

Terminology and interpretation

First, it should be clear that we are *not* talking about *regulatory limits* enforceable through normal regulatory compliance programmes, but rather about *design constraints* which are used as decision criteria for acceptance or rejection of proposed disposal projects. By demonstrating that a repository design meets the design constraints, we hope to ensure to a suitable level of confidence that no member of a future generation will be exposed to an unacceptable radiological dose or to an unacceptable level of risk. Although these criteria and the numerical outcomes of the safety assessment calculations compared to them are measured in the same units as radiological doses, they are not actual radiological doses, but rather estimates of potential exposures that may occur as a result of postulated events and processes in the future. Especially in the distant future, it is widely recognised [Refs. 3, 4] that these calculations should be interpreted as indicators of repository performance and not as measures of health detriment.

In some countries, criteria are expressed in the form of *design targets* rather than design constraints. Whereas a design constraint may be used as a fixed pass/fail criterion for licensing, a design target is only a goal for the design optimisation process. If the target is not met, judgement may be applied during decision making.

In talking about risk criteria, we also need to distinguish between radiological risk, which is actually a conditional risk (conditional on the probability of the scenario giving rise to the exposure), and aggregate risk, which also includes directly in the calculation the probability of the scenario. For normal evolution and other high-probability scenarios, the probability of the exposure is considered to be 1, and radiological risk and aggregate risk are considered to be the same. For low-probability events such as human intrusion into a deep geological repository, when an aggregate risk constraint is used the predicted exposure may be allowed to exceed the normal dose constraint as long as the combined or aggregate risk does not exceed the overall risk constraint (risk aggregation).

Because people are often less willing to accept a high-consequence low probability outcome than a low-consequence high-probability outcome with the same calculated aggregate risk, a single risk criterion may not be appropriate for both situations. For example, risk constraints or targets to be applied to high-consequence events may be more stringent than the risk constraints or targets that would be applied to high-probability or normal evolution scenarios (risk aversion).

Some of the regulatory criteria include some degree of risk aggregation, risk aversion or both. However, most of the following discussion will focus on the criteria used for high-probability normal evolution scenarios. For these scenarios, radiological risk and aggregate risk may be considered to be the same numerically, so that risk criteria and dose criteria can be compared directly to one another with the use of a constant radiological risk conversion factor.

Bases for selection of criteria

Part of the variation in criteria may be attributable to the use of different bases for criteria selection in different countries. Three such proposed bases are: comparison with current radiological protection criteria for operational facilities; comparison with the variability of background radiation exposures; and comparison with generally accepted risk criteria developed without regard for the type of hazard. Of course, despite different philosophical underpinnings, all of these bases are interconnected, and many of the national criteria are justified by comparison with more than one basis, although with differing emphases in different countries.

One approach is the one followed in ICRP-81 [3] and the IAEA Safety Requirements document WS-R-4 [4]. This approach starts from the premise that no person in the future should in the normal course of events receive a dose from the repository any higher than the dose that would be allowed from a nuclear facility today. The dose constraint recommended by both of these documents is 0.3 mSv/a or less. This is the same value as the recommended dose constraint for current practices, which is intended to account for the potential that doses may be received from multiple sources. Some countries have adopted the 0.3 mSv/a dose constraint directly. Some others apply an additional safety factor of two to three to account for additional uncertainties from various sources. This line of argument thus results in dose constraints in the 0.1 mSv/a to 0.3 mSv/a range.

The ICRP also suggests a risk constraint of 10^{-5} per year as being an approximate equivalent to the 0.3 mSv/a dose constraint. Using current values for the risk conversion factor the 0.3 mSv/a dose constraint for high-probability scenarios corresponds to a risk constraint of roughly 2×10^{-5} per year.

A second approach does not depend directly on radiological protection recommendations, but instead compares the additional radiological dose from the normal operation of the repository to the variability of natural background radiation. Since people do not ordinarily take variations in natural background radiation into account when planning everyday activities, it may be considered that an increase in dose in the vicinity of a repository that is small compared to the normal variability should not be of concern. Some countries that have adopted or are considering criteria established on this basis (e.g. Germany and Switzerland) have arrived at a dose criterion of around 0.1 mSv/a. The line of argument used in the US proposals for post-10^4 year doses from the Yucca Mountain Repository rests on a related fundamental basis (comparison with naturally-occurring exposures), although it arrives at a rather different numerical result.

A somewhat different approach from the above two starts from the level of overall risk posed by the repository. A risk target of 10^{-6} per year for the aggregate risk from lower-probability scenarios has been suggested in a number of countries. This one in a million level is sometimes described as a societally acceptable value applicable to a wide variety of risks.

In some countries such as the United Kingdom, this numerical value of risk is used as a design target for normal evolution scenarios. Using current risk conversion factors, this corresponds to a radiological exposure target of 0.015 mSv/a, which is considerably smaller than the constraint values that are arrived at on the basis of radiological protection arguments. In some other cases including the US, a risk constraint of 10^{-5} per year is justified at least partly on grounds of consistency with arguments based on dose limits.

Assessment of conformity

There are several ways in which different interpretations of the assessment of conformity with design criteria can affect the outcome.

First, design criteria can be used in different ways in the optimisation process. If we think of optimisation as a process that affects the design in a range between an upper limit which must not be exceeded (analogous to a 1 mSv/a dose limit for an operating facility) and a lower threshold below which further optimisation would not be considered justified (analogous to a 10 μSv/a de minimis criterion), then we must decide what role the design criterion has: is it the upper limit, the lower threshold, or something in between? In many cases dose and risk constraints are used as upper limits. However, in cases where the numbers are called risk targets rather than limits or constraints (e.g. the UK), the interpretation is closer to the lower threshold. Since the range between upper limits and lower thresholds can, as in the examples quoted, be two orders of magnitude, there is considerable room for variation in the way the criteria are used. Unless we can be certain that criteria being compared are being used in exactly the same way, we cannot be sure whether numerical differences reflect real differences in the level of protection or simply differences in interpretation.

In performing consequence assessments judgments have to be made about choices of parameters and models. Different approaches to assessing the degree of conservatism that is appropriate for such choices can have significant effects on the outcome of the calculation. Only rarely are these approaches quantified; more often, they lie in the realm of professional judgment. The variability that results from these differences could in some cases be larger than the range of variation in the criteria themselves.

Perhaps the largest single component among these "degree of conservatism" judgments is the relative importance given to the conflicting goals of: (i) reflecting as accurately as possible the events and processes leading to the calculated outcome of the safety assessment (e.g. a "design centre" approach) vs. (ii) assuring to as high a level of confidence as possible that the actual exposures to future humans resulting from the existence of the repository will not exceed the calculated values (e.g. a "bounding" approach). The differences between the two extremes can be large, particularly for very long time scales where physical, engineering and computational uncertainties may be large. In general, and reflecting the tension between these conflicting goals, safety assessments lie somewhere in the middle between a true design centre and a completely bounding approach, but differences of this kind between national approaches can be quite substantial, reflecting differing national attitudes towards risk, "safety factors" and the desirable degree of assurance of safety.

Significance of the findings

Overall, radiological protection for disposal involves two components: the setting of criteria (i.e. the definition of acceptable risk) and assessment of conformity with the criteria (i.e. the definition of reasonable assurance). With respect to the first, we are faced here with apparent differences in criteria that appear to range over two orders of magnitude, at least partly as a result of differences between the fundamental bases which are considered most important (radiological protection-based arguments vs.

pure risk arguments vs. comparisons with variability in background). However, these differences must be combined with the differences in the approach to assessment of conformity before we can arrive at a judgment about the comparative level of safety.

The differences in assessment of conformity are less amenable to quantitative comparison than the numerical criteria. Rather than pursuing these differences in assessment in more depth, the LTSC Group concentrated its efforts on other aspects, among them the role of the regulator in the process of decision making [5]; the role of criteria other than radiological protection criteria in regulatory decision making (in addition to other performance criteria, these may also include design criteria and the use of best available techniques); ethical issues revolving around the current generation's obligations to future generations both in the immediate and the distant future; and underlying fundamental safety goals. These studies have profited from close interactions with the IGSC's Timescales Initiative [6] as well as from the input of experts in the social sciences and the humanities. It is intended that the ongoing work of this group will lead to a common understanding of the bases for regulatory criteria in order to facilitate discussion and reach consensus on key aspects, without necessarily attempting to harmonise the criteria themselves.

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Annex 1. **National Criteria for Disposal of Long-Lived Waste in Different Countries**[2]

Country	Target limit of impact (Most exposed individuals)	Other limitations or conditions	Approach to handling of probability or uncertainty	References
Belgium	Dose constraint: 0.1 to 0.3 mSv/yr. Risk constraint: 10^{-5}/yr. (Note: Working values in absence of regulatory values.)	Dose constraint relevant to high probability scenarios and risk constraint to lower probability scenarios.		SAFIR-2.
Canada	Under development: Interim dose constraint of up to 0.3 mSv/yr for design optimisation as recommended by ICRP and IAEA.	Guidance on timescales, institutional control and other indicators is also under development. A public dose criterion of 1 mSv/yr is used for evaluation of human intrusion scenarios.	Under development	
Czech Republic	Dose constraint: 0.25 mSv/yr	Disposal site should provide a natural barrier that assists in keeping the radiological impact to human and the environment within acceptable levels. Safety analysis are required for release scenarios that cannot be excluded	10^{-6}/yr – scenarios with lower probability need not to be considered in the safety analysis	Decree: No. 307/2002 on radiation protection No. 215/1997 on sitting of nuclear facilities
Finland	Dose Limit: 0.1 mSv/yr for normal evolution. For unlikely events, impact assessed against risk equivalent to dose limit.	Release of radionuclides into human environment to be less than nuclide-specific constraints. Dose/risk constraint applies for several thousand years. RN release limitation applies for longer.	Unlikely events assessed quantitatively where practicable, otherwise by qualitative discussion. Deterministic, conservative analyses with assessment of implications of uncertainties.	Govt. Decision: 478/1999. Guide: YVL 8.4.

2. Information courtesy of NEA.

France	Dose Limit: 0.25 mSv/yr for normal evolution.	Dose limit applies for 10^4 yrs, and is a reference for later periods Institutional monitoring assumed to prevent human intrusion before 500 yrs.	Random, unanticipated events subjected to case-by-case judgement, including glaciations after 50 000 years.	RFS III 2f
Germany	In order to provide adequate protection of man and the environment, the criteria define the individual dose as the main safety indicator for the post-closure phase. The analysis has to show that an individual dose limit of 0.3 mSv/a will not be exceeded. (currently under revision)	The Safety Criteria for underground disposal require proof that the site under consideration has favourable mechanical, technical and hydrogeological properties. Safety analysis required for all radionuclide release scenarios that cannot be completely excluded. Demonstration of safety required for period of one million years. Use of further indicators has been required in licensing procedures.	Safety case with uncertainty analyses (requirements during licensing procedures). Presumes knowledge of repository for 500 years, and no human intrusion before then. Targets for individual dose are defined for different classes of likelihood of occurrence. (Derived from natural background radiation variation.) This approach has been chosen, amongst other reasons, in order to avoid conceptual problems linked with the risk concept for long time frames.	Atomic Energy Act of December 23, 1959 (last Amendment April 22, 2002) Safety Criteria for the Final Disposal of Radioactive Wastes in a Mine; 1983,
	Currently, the Safety Criteria for the disposal of radioactive waste are being revised. The revised criteria will take into account recent international developments in waste disposal as well as concerning the structure, content and presentation of the post-closure Safety Case.			
Hungary	Dose Limit: 0.1 mSv/yr. Risk Limit: 10^{-5}/yr, for impact of individual disruptive events.	The consequences of individual disruptive events shall be evaluated using probabilistic analysis.	In probabilistic analysis, events with likelihood of occurrence of less than 10^{-7} event/year may be neglected.	Decree: 47/2003 (VIII.8) E5ZCSM
Japan	(Under development)			
Korea, Rep. of	Dose limit: 0.1 mSv/yr for normal evolution Risk limit : 10-^{6}/yr for probabilistic disruptive events	A public dose criterion of 1 mSv/yr is applied for human intrusion scenarios.	Under development	MOST Notice 2005-17

Netherlands	Dose Limit: 0.1 mSv/yr, (Optimisation goal: 0.04 mSv/yr), for normal evolution.			1st Report, 2003, under Joint Convention on Waste/Spent Fuel.
Norway	(Not available)			
Slovakia	Under development -for radioactive waste that contains significant levels of radionuclides with half-lives greater than 30 years	Dose limit 0.1 mSv/yr. (normal evolution scenarios) and 1 mSv/yr. (intruder scenarios) – for low level and intermediate level radioactive waste with limited content of radionuclides with half-lives greater than 30 years		Decision of Chief Hygienist (1988)
Spain	Dose Limit: 0.1 mSv/yr. Risk Limit: 10^{-6}/yr. Under revision, according to the ICRP 81	Dose limit relevant to high probability scenarios and risk limit to lower probability scenarios. General criteria for site selection		CSN´s Decision on the Proposal of the 1st General Radioactive Waste Plan, approved in 1997. CSN Report to Parliament, 2nd semester 1985. 1st Report, 2003, under Joint Convention on Waste/Spent Fuel.

Sweden	Risk Limit: 10^{-6}/yr. (Dose/risk conversion factor of 0.073 Sv^{-1} to be used.)	Biodiversity and biological resources also to be protected against the effects of ionising radiation. Quantitative assessment, including collective dose, to be made for the first 1 000 yrs. For period beyond 1 000yrs, general consideration of various possible scenarios for evolution of the repository's properties, its environment and the biosphere (SSI). A safety assessment shall comprise as long time as barrier functions are required, but at least 10 000 years.	Uncertainties in the description of the functions, scenarios, calculation models and calculation parameters used in the description as well as how variations in barrier properties have been handled in the safety assessment must be reported, including the reporting of a sensitivity analysis which shows how the uncertainties affect the description of barrier performance and the analysis of consequences to human health and the environment	SSI FS 1998:1 SSI FS 2005:5 SKI FS 2002:1
Switzerland	Dose Constraint: 0.1 mSv/yr. Risk Target: 10^{-6}/yr.	Dose constraint relevant to high probability scenarios and risk target to lower probability scenarios. (Valid for all time.) Complete containment for 1 000 years.	For long-term dose calculations: - reference biospheres - population with realistic habits - conservative assumptions	HSK R-21
United Kingdom	Dose constraint: 0.3 mSv/yr. Risk target: $<10^{-6}$/yr. (Dose/risk conversion factor of 0.06 per Sv to be used for dose-rates less than 0.5 Sv/a)	Dose constraint applies to period before control is withdrawn. Risk target to longer periods Required to show that radionuclide releases are unlikely to lead to significant increase in levels of radioactivity in the accessible environment.	Presentation of information on risks to include desegregation of probability and consequences, where practicable.	Environment Agency "GRA" Document, 1997 (EA, SEPA, DoE(NI)).

USA (Yucca Mountain)[3]	Dose Limit (no human intrusion): 0.15 mSv/yr. (Equivalent to fatal cancer risk of $8.5*10^{-6}$/yr using conversion factor of 0.0575 cancers per Sv). Dose Limit (after human intrusion): 0.15 mSv/yr as result of a human intrusion at or before 10^4 yrs after disposal.	Detailed restrictions apply for 10^4 yrs to radionuclide concentrations in groundwater. Compliance with quantitative dose limit required for 10^4 yrs. Requirement to calculate peak dose if it occurs later, (up to 10^6 yrs, i.e. the assumed limit of geologic stability), but the quantitative standard does not apply beyond 10^4 yrs.	10^{-8}/yr cut off for consideration of events/scenarios. (Corresponds to≈ 10^{-4}/10 000 yrs for post-closure period.)	40 CFR Part 197, as implemented in 10 CFR Part 63
IAEA	Dose constraint: 0.3 mSv/yr. Risk constraint: 10^{-5}/yr.		Multiple lines of reasoning, e.g. based on natural analogues and paleohydrological studies of site and host rock	Safety Requirements currently in draft.

3. In 2005, certain changes were proposed to the Yucca Mountain standards at 40 CFR Part 197. These changes would extend the period over which a quantitative dose limit applies, out to the estimated time of geologic stability at Yucca Mountain, approximately 1 million years. The dose limits for the first 10 000 years after disposal would remain as shown in the table. The proposed rule would establish a new dose limit of 3.5 mSv/year for the period from 10 000 years to 1 million years for undisturbed performance and, separately, in the event of human intrusion. These limits would assure that any people living near Yucca Mountain up to 1 million years in the future would not receive total radiation doses that exceed natural background radiation levels in comparable geographic and geologic regions. The groundwater standard would not extend beyond 10 000 years. For more details on the proposed rule, visit: www.epa.gov/radiation/yucca. The changes have not yet been made final.

HOW HAS THE SAFETY CASE EVOLVED IN THE SWISS NATIONAL PROGRAMME?

P. Zuidema, J. Schneider
Nagra National Co-operative for the Disposal of Radioactive Waste, Switzerland

Introduction

In Switzerland, nuclear power production by the current five plants contributes approximately 40% of the electricity consumed; the other 60% come from hydro power. The first commercial nuclear power plant went into operation in 1969. In 1972, the nuclear power plant operators and the Swiss Federal Government – which is responsible for waste arising from medicine, industry and research – founded the National Co-operative for the Disposal of Radioactive Waste (Nagra). Nagra has been charged by the producers of radioactive waste with the responsibility for preparing and implementing sustainable waste management solutions in Switzerland. Accordingly, since the late 70s investigations have been underway with respect to the safe disposal of radioactive wastes in Switzerland, both for low- and intermediate-level waste (L/ILW) and for high-level waste (HLW); see the broad review of options reported in 1978 [1].

In the course of a large number of projects that Nagra has developed in the meantime, Nagra has gained broad experience in compiling safety cases. In order to illustrate how the safety case (concept, methodology and application) has evolved in the Swiss national programme during the time period from the late 70s up to now, the present paper is organised as follows: First, to set the scene, an overview of the Swiss waste management programme is presented, and the important milestones are highlighted (Section 2). This is followed by a section on the evolution of the legal and regulatory framework in that time period (Section 3). Next, the nature of the safety case is discussed and factors to consider in illustrating the evolution of the safety case concept are listed. Making use of "example safety cases" compiled at various points in time within the considered time period, with these factors it is shown how the safety cases have evolved in that period (Section 4). Section 5 presents a summary and conclusions.

Overview of the Swiss waste management programme

A first important milestone in the Swiss waste management programme was the publication of the review of options report in 1978 [1]. In 1980, the crystalline basement was allocated first priority for the HLW programme, and a regional characterisation programme was initiated. In 1981 a report on the technical basis and the strategy for site selection for a L/ILW repository was published; this report included a list of 100 potential sites [2]; this report was followed by a report that narrowed the list of sites down to 20 sites. In 1985 a further milestone was reached with submission of Project Gewähr [3-11], which aimed at demonstrating disposal feasibility of L/ILW and HLW. The host rock selected for this purpose was marl for L/ILW and the crystalline basement of northern Switzerland for HLW. The evaluation of feasibility for HLW was based on the results of the regional field programme mentioned above (deep boreholes, 2-D reflection seismics, investigation of regional geology, etc.), lab work and studies. In their review, which was published in 1988, the Swiss authorities came to the following conclusions: (i) that disposal feasibility for L/ILW had been demonstrated; (ii) that for

HLW, long-term safety and engineering feasibility had been adequately demonstrated but that siting feasibility – the likelihood that a sufficiently large block of crystalline rock with the properties assumed in the study could be found and characterised with sufficient reliability – was not yet fully convincing due to the complex tectonic situation in northern Switzerland. The authorities also required that the investigations within the HLW programme should be extended to sedimentary formations. In the period following this decision, three main lines of activities were pursued; one within the L/ILW programme and two within the HLW programme. In the remainder of this paper, we focus on the HLW programme.

After Project Gewähr, additional regional geological investigations were carried out. Overall, the investigation programme of the crystalline basement of northern Switzerland was conducted during 1981-1993 and included seven deep boreholes with depths between about 1 300 and 2 500 m, about 400 km of reflection seismics and a large number of additional investigations. All this information was compiled in a geological synthesis report that was published in 1994 [12]. A safety report making use of this information, labelled "Kristallin-I", was published in the same year [13].

Already in the field programme for the crystalline basement, Nagra had characterised the most promising sedimentary layers overlying the crystalline basement. From this data, together with data from other sources and taking into account the good general understanding of the geology of Switzerland, Nagra published a first interim report on the broad geological possibilities in sedimentary rocks in 1988 [14]. Based on this interim report and two other interim reports [15,16], in 1994 an agreement with the relevant Swiss authorities was reached that field investigations for the HLW programme should focus on the Opalinus Clay (a Jurassic claystone formation) as a host rock and the Zürcher Weinland as a potential siting region for a comprehensive demonstration of disposal feasibility. After extensive investigations (including a borehole and 3-D seismics at the site, regional geological investigations and studies, experiments at the Mont Terri rock laboratory, and using information from other sources), Nagra submitted at the end of 2002 comprehensive project documentation on the feasibility of safe disposal of spent fuel (SF), vitrified HLW from reprocessing of SF and long-lived intermediate-level waste (ILW) in Opalinus Clay in the potential siting region of the Zürcher Weinland (Project Entsorgungsnachweis [17-19]). The recently published reviews of the Swiss safety authorities and their experts, as well as the review by an international review team under the auspices of the OECD/NEA, which was published in April 2004, all came to a positive conclusion about the project [20-23]. The review phase was followed by a broad, three-month public consultation phase in the fourth quarter of 2005. Based on the results of the review and the public consultation phase [24], the Swiss Government (the Federal Council) concluded on 28 June 2006 that disposal feasibility of SF/HLW/ILW in Switzerland had been successfully demonstrated [25].

Legal and regulatory framework

In Switzerland, the management of radioactive waste is regulated by a legal framework consistent with the requirements of the IAEA Joint Convention on the Safety of Spent Fuel and on the Safety of Radioactive Waste Management [26]. At the highest level, the Federal Constitution (Art. 90) states that atomic energy legislation is a federal matter.

As early as 1959, the Atomic Law (as it was called then) entered into force (AtG, [27]). It formed the legal basis on which the management of radioactive waste arising from the peaceful use of nuclear energy (i.e. from the operation and decommissioning of nuclear power plants) was founded. The Law was supplemented by the Federal Government Act on the Atomic Law of 1978 (BB/AtG, [28]), which embodies the principle that the producers of radioactive waste are responsible for its safe disposal and all associated costs. The AtG and the BB/AtG also defined the different licences required in the step-wise approach to repository implementation. The Atomic Law was recently revised; the revised version, which entered into force in 2005, is termed "Nuclear Energy Law" ("Kernenergiegesetz" or, abbreviated, "KEG" in German, [29]). The law is accompanied by a corresponding ordinance that entered into force at the same time [30]. The law includes, in addition to the above points, the following key waste management related elements: (i) decisions on the selection of a specific site for a deep geological repository (the general licence) are subject to an optional national referendum ("fakultatives nationales Referendum"); (ii) there is a 10-year moratorium on reprocessing of spent fuel starting on July 1, 2006; (iii) all radioactive waste (i.e. both L/ILW and HLW) has to be disposed of in a deep geological repository that has to allow for monitoring for an extended period of time before full closure; during this period, retrieval of the waste should be feasible with reasonable effort; (iv) a site selection procedure has to be defined by the Federal Government ("Sectoral Plan").

The Swiss Federal Nuclear Safety Inspectorate (HSK) is the supervisory authority. In HSK's guideline HSK-R-21, the protection objectives for disposal of radioactive waste are defined [31]. HSK-R-21 was first published in 1980; it was revised in 1993 and is currently again under revision. It is beyond the scope of this paper to discuss in detail the contents of HSK-R-21; however, in order to illustrate the evolution between 1980 and 1993, a number of key differences between the original version and the currently applicable version are listed below.

- *Aims of deep geological disposal – 1980:* implicit in introduction; *1993:* clearly stated.
- *Principles for deep geological disposal – 1980:* missing; *1993:* 6 principles formulated.
- *Protection objectives – 1980:* two protection objectives (Protection objective 1: dose limit of 10 mrem/a; Protection objective 2: after closure, no measures shall be necessary to provide safety; repository must be designed so that it can be closed within a few years.); *1993:* three protection objectives (Protection objective 1: dose limit of 0.1 mSv/a; Protection objective 2: individual radiological risk of fatality from a sealed repository subsequent upon unlikely processes and events not taken into consideration in Protection objective 1 shall at no time exceed one in a million per year; Protection objective 3: after closure, no measures shall be necessary to provide safety; repository must be designed so that it can be closed within a few years.
- *Explanatory comments on safety analysis – 1980:* 3 topics: (i) events, (ii) assumptions for the calculations, (iii) treatment of uncertainties; *1993:* 7 topics: (i) predictive modelling, (ii) predictions into the distant future, (iii) site selection and investigation, (iv) processes and events, (v) affected population group, (vi) models and data, (vii) safety enhancing measures.
- *Definitions – 1980:* 4 definitions: (i) closure, (ii) dose, (iii) final disposal, (iv) repository; *1993:* 12 definitions: (i) closure, (ii) conservative, (iii) disposal, (iv) dose, (v) repository system, (vi) host rock, (vii) long-term safety, (viii) multi-barrier concept, (ix) risk, (x) scenario, (xi) validation, (xii) verification.

Evolution of the safety case in the Swiss national programme

In this section three selected projects introduced in Section 2 will be used as examples to illustrate specific aspects of the evolution of the safety case in the Swiss national programme. These are: (i) Project Gewähr – HLW part (PG, 1985); (ii) Kristallin-I (Kri-I, 1994); (iii) Project Entsorgungsnachweis (PE, 2002). The purpose of all three studies was to evaluate siting feasibility; and all were based on field data obtained for this purpose.

Before comparing these studies, our understanding of the nature of a safety case is briefly discussed here. It is the purpose of a safety case to provide a systematic integration of the scientific information relevant for the chosen system (i.e. site/host rock and design) and its evaluation in terms of its meaning with respect to safety. Key elements of a safety case are thus: (i) the system (i.e. site/host rock and design); (ii) the underlying scientific basis; (iii) the team for the interdisciplinary integration and evaluation of the information; (iv) a well-defined, systematic approach (the methodology); (v) an adequate "tool box" (methods, codes, computers, etc.) to conduct the work; (vi) traceable and transparent documentation.

Apart from the legal and regulatory framework, which was discussed in Section 3, the following factors may be usefully considered in illustrating the evolution of the safety case concept:

Scope of the safety case

- **System –** In all three safety reports the total system consisting of the near field (waste form, canister, bentonite), the geosphere and the biosphere was considered. In the HLW part of PG, the focus was on vitrified HLW from reprocessing of SF. However, in an appendix in [7], the option of direct disposal of SF was also briefly discussed, and illustrative calculations on post-closure safety were presented. In Kri-I, only vitrified HLW was considered; but in conjunction with that study, some preliminary work on safety-related aspects of repositories for SF was performed [32-34]. In PE, SF, HLW and ILW were treated with equal weight.
- **Range of scenarios and assessment cases –** In all three safety reports, a wide range of scenarios are considered; some of which are discussed qualitatively, while others are analysed numerically by defining and evaluating so-called assessment cases (i.e. cases for which the consequences in terms of dose are analysed numerically). We note that the number of assessment cases, as well as the level of detail considered in the assessment cases, increased from PG to Kri-I to PE; the main reasons being the broader and improved scientific basis and the increased availability of computing power.
- **Level of detail considered in calculations –** The models and codes used were refined from one project to the next so that in the calculations for PE, generally a higher level of detail was considered than in PG. Also, there was a corresponding increase in breadth and depth of the data used from PG to Kri-I to PE, reflecting the considerable efforts spent in improving the scientific basis and in obtaining specific datasets (including geological data from field investigations and geochemical data based on lab measurements) during the time interval between PG and PE. This will be discussed further (below).
- **Scope and depth of argumentation –** Generally, the scope and depth of argumentation increased from PG to Kri-I to PE, in line with the evolution of the concept of the safety case internationally, where one key feature is the increased relative weight of arguments compared to calculation results. As an example, in PE the concept of safety functions was used for the first time as a central part of the argumentation.

Developing a safety case

- **The team and the interactions between the different management functions –** The integration and evaluation of the available information in a safety case is an interdisciplinary task that requires a well-trained team of highly qualified scientists that also have a high degree of social competence. The building up of such a team takes time. Ideally, a majority of the members should be involved in the compilation of several major projects. This, in turn, is only possible within a company with a "culture" that encourages qualified personnel to stay with the company for considerable periods of time. The team and the interactions between the different management functions was not explicitly discussed in PG or in Kri-I, while in PE this topic was treated prominently (Chapter 3 & Appendix 4 in [19]).

- **Safety case methodology –** The safety case methodology evolved considerably from PG to Kri-I to PE. In PG, the closest to a description of the safety case methodology is a short chapter on the approach to conducting a safety analysis (Chapter 3 in [7]). In Kri-I, the corresponding chapter (Chapter 2 in [13]) takes up about twice as much space, and the term "safety case" is used for the first time, although only in one specific instance in the context of "building a robust safety case". In contrast, in PE, there is a dedicated chapter (Chapter 3 in [19]) with the title "Methodology for developing the safety case"; and it is shown how the methodology is developed based on five clearly defined assessment principles. One important aspect of the safety case methodology is the use and management of FEPs (features, events or processes). This, too, has evolved substantially from PG to Kri-I to PE (see discussion below under "Scenario development & the identification of assessment cases"). Also, we note that the concept of "safety functions" was used for the first time in PE.

- **Scenario development & the identification of assessment cases –** In both PG and Kri-I, the "conventional approach" to scenario development and the identification of assessment cases was chosen, where the first step consists in the development of an extensive, project-specific FEP database (see Figure 4.1.1 in [13]). In PE, an alternative approach was used, where the project-specific FEP database was developed as a separate activity and was used primarily as a "book-keeping tool" to ensure phenomenological completeness of the assessment; i.e. to show that a thorough identification had been made of all the FEPs that could affect long-term safety, and that these FEPs had been adequately treated in safety assessment (see Figure A4.1 in [19]). In PE, and in contrast to PG and Kri-I, the assessment cases were derived as a result of a careful evaluation of the scientific basis, guided by a wide range of "insight calculations" and sensitivity analyses. Also, in PE an international FEP database was used for a "completeness check" of the project-specific FEP database. This is discussed in more detail in a separate "FEP management report" for PE [36].

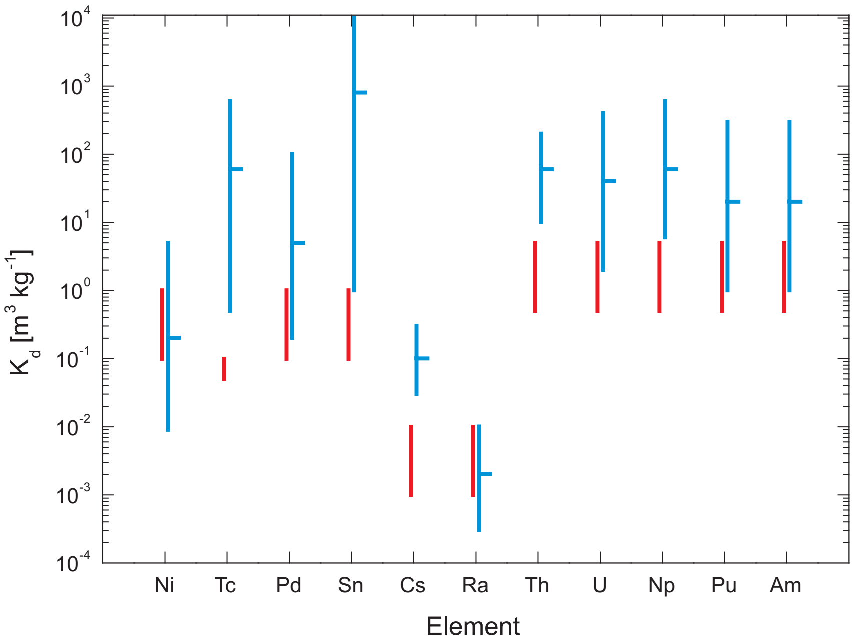
- **Description of the system and its possible evolutions –** In PG, the system and its possible evolutions are described in a report with the title "the system of the safety barriers" [6]. In Kri-I, a separate chapter (Chapter 3) is dedicated to the description of the Kristallin-I disposal system. In PE, one chapter (Chapter 4) describes the disposal system as implemented, and a separate chapter (Chapter 5) discusses system evolution. When comparing the three projects, one notes that in PG, each system component (and processes that could affect its evolution) is first described; this is immediately followed by a section on the corresponding modelling approach used in the safety assessment. In contrast, in Kri-I, the description of the system (and processes that could affect its evolution) and how the system and its evolution is modelled (Chapter 5) are discussed in separate chapters. In PE, a distinction is made between system description (and a description of its evolution) and *the evaluation of the performance of the disposal system* (Chapter 7). This reflects a trend to clearly distinguish between the description of the system and its evolution on one hand and the approach to evaluating the performance of the system (for which modelling is "only" a tool, rather than an aim in itself) on the other hand. Consequently, and in contrast to PG and Kri-I, in PE, the details of the models are described in a separate report [35]. Another important point in the context of the description of the system and its evolution is the explorability and predictability of the site. In fact, one of the major comments in the authorities' review of PG was related to this topic, as discussed in the introduction to Section 2. Also, it can be argued that one of the key factors in the success of PE was the excellent explorability and predictability of the sediments (including the host rock Opalinus Clay) in the Zürcher Weinland.

- **Sensitivity analysis and the treatment of uncertainties –** In all three projects sensitivity analyses were performed. However, we note that the way the corresponding results are presented has evolved from PG to Kri-I to PE. In PG, although a number of sensitivity analyses were performed, the term as such was not used. In Kri-I, sensitivity analyses were used extensively, though not always under this heading (see, e.g. Figure 5.3.8 ff where the maximum calculated dose due to a given nuclide is plotted against a large number of representations of geosphere transport pathways). In PE, there is a special section dedicated to this topic (Section 6.7), where the purpose of performing sensitivity analyses is explained explicitly. Also, in PE, a range of simplified so-called "insight models" are used to increase system understanding. In these insight calculations, parameters are varied over a wide range (well beyond that supported by observations) to illustrate system behaviour even under extreme assumptions. One example is the illustration of the effect of hypothetical water-conducting discontinuities in the Opalinus Clay (Figure 6.7-1 in [19]). Generally, the importance of a systematic treatment of uncertainties has increased in the time span between PG and PE. This is in agreement with the increased weight that the authorities have given to this point, as evidenced, for example, by the introduction of a risk-based protection objective in the revised regulatory guideline in 1993 (see Section 3). As a consequence, probabilistic analyses (complementary to the deterministic analyses) were introduced in PE.

- **Analysis of assessment cases –** Under this header three aspects of the analysis of assessment cases are considered: (i) the source, breadth and depth of input data; (ii) the types of codes used; and (iii) to what extent a methodology was defined and applied to ensure that all relevant scientific understanding is taken into account in an appropriate manner in the definition and evaluation of assessment cases ("bias audit"). **(i) Source, breadth and depth of input data:** Generally, the breadth and depth of the input data increased over time. One example is the set of sorption values for bentonite. In PG and Kri-I, the emphasis was on literature reviews, from which "realistic" and "conservative" values were derived. In PE, the emphasis shifted towards lab measurements for key elements conducted under controlled conditions that were adapted specifically to that project, and the values were provided in terms of reference case, lower limit and upper limit values. Furthermore, and in contrast to the earlier projects, if unusually high values were measured and a critical evaluation showed that they could in fact be supported, these were then used to derive the reference case values; whereas in earlier projects such measurements were often discarded as being "too optimistic". This is illustrated in Figure 1. **(ii) Types of codes used:** We can distinguish between three "levels" or types of codes: (i) Codes used to calculate dose or risk ("PA codes"); (ii) simplified "insight models" (see previous bullet point); (iii) detailed "process models" (typically used to build up process understanding for specific processes and/or to obtain corresponding input parameters for the PA codes). In PE, this distinction was made in a rigorous manner and broadly applied. Already in Kri-I, wide use was made of insight models. Generally, we note that the codes used today are much more user-friendly than in earlier times (PG), that they typically require much less computing time and that the level of detail that they consider has increased, in some instances dramatically. For PG, detailed planning of the use of external computing resources was essential; while for PE, all assessment cases were evaluated "in house" by a single contractor within a relatively short time period. Also, in PE, probabilistic analyses were used for the first time (complementary to the deterministic analyses). **(iii) Methodology used to ensure that all relevant scientific understanding is taken into account in an appropriate manner in the definition and evaluation of assessment cases ("bias audit"):** PG and Kri-I: Not specifically addressed. PE: This is a theme that was always present when developing PE. Consequently, this is addressed in a number of places throughout the documentation of PE [Chapter 3 and Appendix 4 in [19], in the Models, Codes and Data Report [35] where after the description of each model a table shows how the scientific understanding is represented in that model, and in the FEP Management Report [36], where it is shown in a systematic way how the scientific understanding is represented in analysing the assessment cases (see, e.g. Table A6.1.1)].

- **Use of arguments –** In line with the evolution of the concept of the safety case internationally, where one key feature is the increased relative weight of arguments compared to calculation results, the use of arguments in the development of the safety case has increased noticeably from PG to Kri-I to PE. One example is the concept of so-called “reserve FEPs”; i.e. features, events or processes that are neglected in the analysis of assessment cases, but that are known to be beneficial to safety. In PG, this line of argument was not used. It was introduced for the first time during the Kri-I project, and was used again as one of the arguments for safety in PE. Other examples include the use of a number of alternative safety indicators (other than dose or risk) and the use of natural tracer profiles, both of which were used extensively in PE.
- **Statement of confidence in results and indication of how remaining uncertainties can be resolved –** This topic was brought up in discussions about the safety case in international fora after the submission of PG and Kri-I. Consequently this is addressed for the first time in PE.
- **Quality management –** The importance of explicitly addressing quality management has increased between PG and PE. In the documentation of PE, a special section is dedicated to this topic (Appendix 8 in [35]).

Figure 1. **Sorption values for bentonite used in Kri-I (red bars without reference case values, from Table 3.7.3 in [13]) and in PE (blue bars with reference case values, from Table A2.6 in [19]).**



Structure of the safety case

- **Reporting structure & reference reports –** PG is documented in 8 main technical reports: a concept and overview report, a report on the properties of radioactive waste and its allocation to the two types of repositories (L/ILW and HLW), and, for each of these two types of repositories, a report on construction and operation, on the safety barrier system and on long-term safety (Figure 1-1 in [11]). Kri-I comprises as key technical reports a geosynthesis taking into account new geological information that had been obtained after PG had been submitted [12] and a re-evaluation of long-term safety of a repository for vitrified HLW [13]. In PE, at the highest level, there are three key technical project reports [17-19]. The three project reports, in turn, are backed up by more detailed technical “reference reports” (Figure 1.4-1 in [19]). The structure reflects the purpose of the three projects: PG aimed at demonstrating disposal feasibility for L/ILW on one hand and for SF, HLW and long-lived ILW on the other hand. PE, as a “left-over” from PG, aimed at demonstrating disposal feasibility for SF, HLW and long-lived ILW. The main purpose of Kri-I, in contrast, was to document progress made since PG, both in the understanding of the crystalline basement of northern Switzerland and in the approach to evaluating long-term safety of a repository for HLW in that host rock. We also note that in PE, in accordance with the explicitly stated principle of “transparency and traceability” (Table 2.6-2 in [19]), the safety report is divided into two parts: (i) the main safety report [19] and (ii) a report with all the details on models, codes and data [35] that would allow anyone to independently re-calculate all results presented in the main safety report.

- **Structure of the safety report –** When comparing the structure of the safety reports from PG to Kri-I to PE, it is evident that it reflects the evolution of the safety case concept internationally. For example, in PG and in Kri-I there is a lot of emphasis on the results of calculated doses, while in PE, the main emphasis is on the set of arguments used to build the safety case, in accordance with recent definitions of “safety case” (see, e.g. [37]). Other noteworthy points include the first-time use, in PE, of a flowchart for the development of the safety case that is used as a guide through the safety report (Figure 1.5-1 in [19]) and the use of summary tables at the end of chapters that serve as input to the following chapters (e.g. Tables 5.7-1, 6.8-1 in [19]).

Summary and conclusions

In this paper the evolution of the safety case in the Swiss national programme has been discussed and illustrated with a number of specific examples making use of “example safety cases” compiled at various points in time. These example safety cases are (i) Project Gewähr – HLW part (PG, 1985), (ii) the Kristallin-I study (Kri-I, 1994) and (iii) Project Entsorgungsnachweis (PE, 2002). The key findings may be summarised as follows:

- **Importance of the scientific basis –** A well-supported scientific basis is essential for a convincing safety case. Typically, the scientific basis increases in breadth as well as in depth as a project moves from one stage to the next. The reliability of the scientific basis depends strongly on the system chosen (site/host rock and design), since both the explorability (i.e. the reliability with which the geological environment of the repository can be characterised) and the predictability (i.e. the reliability with which the temporal evolution of the system can be described) are largely determined by the choice of the system. The compilation and evaluation of safety cases for repository concepts first in the crystalline basement of northern Switzerland and then in the Opalinus Clay in the Zürcher Weinland clearly showed the importance of good explorability in providing confidence in the performance of the geosphere barrier. For the Opalinus Clay (which has homogeneous properties over large scales and is suitable for seismic investigations), explorability has proven to be better than for the crystalline basement (which is in northern Switzerland highly heterogeneous and – due to the sedimentary cover – is less suitable for seismics investigations). In fact, a large part of the increase in the level of confidence in the conclusions regarding the role of the geosphere barrier in the Swiss HLW programme from PG to PE may be traced back to the qualities of Opalinus Clay as discussed above.
- **Importance of the team –** The compilation of the scientific basis and its proper use in building the safety case involves a large range of different disciplines, each with their own jargon, that have to work together efficiently and effectively. This requires an experienced

team with members that are both good scientists and have proven social skills. Furthermore, members should be able to communicate not only their own views on specific topics, but also those of the wider scientific community as a whole, in order to ensure an adequate representation of science in the safety case. The building of such a team, in turn, can only be successful in companies with a culture that encourages long-term commitment of the personnel.

- **Importance of arguments –** One important development in the safety case is the increased importance of arguments. These arguments should highlight both the quality of the system chosen and the quality of the underlying scientific basis, since a safety case is only convincing if both of these prerequisites are met. One example for such an argument is the use of independent evidence; e.g., in PE, the use of measured isotope profiles to support the statement that in Opalinus Clay over the relevant timescales and distances, diffusion has been, and will be, the dominant transport mechanism.
- **Importance of the methodology –** In order to first build, and then make the best possible use of, the scientific basis in compiling a safety case, an adequate and well-defined methodology is needed. In PE, a considerable effort was spent on this, paying special attention to the interaction of the different groups involved in the work, and to the proper inclusion of science. One example here is the involvement of independent experts from outside the waste management community (e.g. from the oil and gas industry) in building up system understanding for the Opalinus Clay host rock. Also, a new way of FEP management was introduced for PE, with emphasis on ensuring adequate use of the scientific basis in the safety case.
- **Importance of the documentation –** The documentation of the safety case should meet two potentially conflicting requirements: it should be both transparent and traceable. The first of these refers to clarity and readability, the second to retrievability of all relevant information that should allow, in principle, an independent evaluation/re-calculation of all presented results. In PE, this was addressed by separating the documentation into main arguments for the safety case (→ safety report) and a detailed description of the models, codes and data used (→ models, codes and data report), as well as a large number of technical reference reports.

Thus, the safety case has undergone an evolution, rather than a revolution, in the time period since the first major projects were worked out, involving a broadening and deepening of the arguments and analyses. This is well reflected in a corresponding change in wording: In the early days, the term "safety analysis" was often used; today, it is "safety case".

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PROPOSED AMENDMENTS TO THE ENVIRONMENTAL RADIATION PROTECTION STANDARDS FOR YUCCA MOUNTAIN, NEVADA

B. Gitlin, R.L. Clark
U.S. Environmental Protection Agency

Abstract

The Environmental Protection Agency (EPA) proposed amendments to its radiation protection standards for the potential spent nuclear fuel and high-level radioactive waste disposal system in Yucca Mountain, Nevada on 22 August 2005. The original standards are found in Part 197 of Title 40 of the Code of Federal Regulations (40 CFR Part 197). The Energy Policy Act of 1992 directed, and gave the authority to, EPA to take this action based upon input from the National Academy of Sciences (NAS). The final original standards were published in the *Federal Register* (66 FR 32073) on 13 June 2001. In July 2004, a Federal court remanded part of the standards to EPA for reconsideration.

The 40 CFR Part 197 standards, as issued in 2001, have four major parts: (1) individual-protection during storage activities; (2) individual-protection following closure of the repository; (3) human-intrusion; and (4) ground-water protection. The storage standard is 150 microsieverts (µSv) (15 mrem) annual committed effective dose equivalent (CEDE) to any member of the general public. The disposal standards are: (1) 150 µSv (15 mrem) annual CEDE for the reasonably maximally exposed individual (RMEI) for 10 000 years after disposal; (2) 150 µSv (15 mrem) annual CEDE received by the RMEI within 10 000 years after disposal as a result of human intrusion; and (3) the levels of radionuclides in the ground water cannot cause annual individual doses to exceed: (1) 40 µSv (4 mrem) per year from beta and gamma emitters or (2) 5 picocuries per liter (pCi/L) of radium-226 and -228 or 15 pCi/L of gross alpha activity. There were also requirements related to the post–10 000-year period, the basis of compliance judgments, and performance assessments.

The Agency's proposed amendments would retain the individual-protection standard established in the 2001 standards, up to 10 000 years. In addition, the compliance period for the individual-protection and human-intrusion standards would be increased to 1 million years and the annual CEDE limit between 10 000 and 1 million years would be 3.5 mSv (350 mrem). There are also proposed requirements for the way performance assessments will be conducted. Finally, the dose calculation methodology would be updated to an ICRP 60 and 72 basis instead of ICRP 26 and 30.

The comment period on the proposed amendments ended 21 November 2005. The Agency is analyzing the comments and will publish its responses when issuing the final standards. The proposed standards and the support documents are available at http://www.epa.gov/radiation/yucca/index.html. The docket containing all of the comments is under Docket ID EPA-HQ-OAR-2005-0083 at: http://www.regulations.gov.

Introduction

The U.S. Environmental Protection Agency (EPA) first issued radiation protection standards for the potential spent nuclear fuel and high-level radioactive waste disposal system in Yucca Mountain, Nevada on 13 June 2001 (the 2001 standards [1]) under the authority of the Energy Policy Act of 1992 (EnPA [2]). (The term "repository" is used in this paper to refer to the mined facility, while the term "disposal system" is used to refer to the entirety of the mined facility, the engineered barriers, and the geologic barrier.) The EnPA also directed EPA to set the standards "based upon and consistent with" the results of a study by the National Academy of Sciences (NAS) "to provide [to EPA]...findings and recommendations on reasonable standards for protection of the public health and safety…." (the NAS Report [3]). The standards are in Part 197 of Title 40 of the Code of Federal Regulations (40 CFR Part 197).

After the standards were issued, petitions for review were filed in Federal courts by the State of Nevada, several environmental and public interest groups led by the Natural Resources Defense Council, and the Nuclear Energy Institute. The standards survived every challenge except one regarding the compliance period. The Court ruled that the 10 000-year compliance period was not based upon and consistent with a recommendation in the NAS Report [3]. The NAS recommendation was:

> "…there is no scientific reason for limiting the time period of an individual-risk standard in this way (10 000 years). We believe that compliance assessment is feasible for most physical and geologic aspects of repository performance on the time scale of the long-term stability of the fundamental geologic regimes – a time scale that is on the order of 10^6 years at Yucca Mountain – and that at least some potentially important exposures might not occur until after several hundred thousand years. For these reasons, we recommend that compliance assessment be conducted for the time when the greatest risk occurs, within the limits imposed by long-term stability of the geologic environment." [3]

Notably, NAS also said: "Nevertheless, we note that although the selection of a time period of applicability has scientific elements, it also has policy aspects that we have not addressed. For example, EPA might choose to establish consistent policies for managing risks from disposal of both long-lived hazardous non-radioactive materials and radioactive materials." The Agency's longest-term disposal standards and regulations for both non-radioactive and radioactive hazardous wastes extended only to 10 000 years. Despite EPA's explanations of those factors, the Court ruled that EPA's compliance period for Yucca Mountain was not based upon and consistent with the NAS recommendation and that EPA had not sufficiently justified its decision to set the 10 000-year compliance period on policy grounds.

On 22 August 2005, the Agency proposed amendments to address the Court ruling [4]. The parts of the standards not affecting the extension of the compliance period are not being proposed for change, with the exception of updating the dose methodology. Thus, changes were not proposed to the storage standards, the characteristics of the reasonably maximally exposed individual, and the ground-water protection standards, for example. The comment period ended 21 November 2005. Hearings were held in early October 2005 in Amargosa Valley and Las Vegas, Nevada, and Washington D.C.

In previous papers, EPA has reported the findings and recommendations in the NAS Report, public comments received from the review of the NAS Report, the considerations made while establishing the 2001 standards, and the contents of those standards. This paper discusses the proposed amendments to the 2001 standards.

Overview of the 2001 Disposal Standards

Subpart B of 40 CFR Part 197 contains the disposal standards for: (a) protection of individuals; (b) human intrusion; and (c) ground-water protection. The disposal phase is considered to start when the repository is closed. Disposal was the subject of the findings and recommendations of the NAS Report [3].

Individual-protection Standard. The individual-protection standard is 150 µSv (15 mrem) committed effective dose equivalent (CEDE) per year for 10,000 years after closure. The Agency uses the dose incurred by a reasonably maximally exposed individual (RMEI) to compare with the dose limits. The concept is similar to the critical group approach in that its purpose is to project doses that are among the highest but still in a reasonably expected range rather than the highest theoretical dose. The location of the RMEI must be assumed to be in the accessible environment above the point of highest concentration of radionuclides in the aquifer. The accessible environment can be no farther down-gradient than the southern edge of the Nevada Test Site (NTS), or about 18 kilometers south of the repository

Ground-water Protection Standards. These standards provide separate protection of ground water. The overall goal is to prevent adverse effects upon human health and the environment by preventing contamination rather than relying upon later mitigation. The limits are the same as the maximum contaminant levels for radionuclides under the Safe Drinking Water Act. The compliance period for these standards is 10 000 years based upon undisturbed performance, i.e. the assumption that the repository is not affected by human intrusion or unlikely features, events, or processes (FEPs).

Human-intrusion Standard. The human-intrusion standard is 150 µSv (15 mrem) CEDE per year for 10 000 years after closure. The required human-intrusion scenario is a single intrusion as a result of exploratory drilling for ground water. The EPA specifies certain borehole parameters that DOE must use to assess the dose received by the RMEI as a result of releases that travel through the borehole, without including the effects of unlikely FEPs. The timing of the intrusion is to be established by NRC based upon the earliest time that current technology and practices could lead to waste package penetration without the drillers noticing it. However, it must not occur sooner than the cessation of active institutional controls. Finally, the standard requires that the human-intrusion analysis be done using the same assumptions and RMEI characteristics as those required for the individual-protection standard.

Proposed Amendments to the 2001 Standards

Scope of the rulemaking

The proposed rulemaking was limited to those portions of the 2001 standards that were affected by the court ruling, i.e. the compliance period for the individual-protection and the human-intrusion standards and certain supporting items. Even though the ground-water protection standards also have a 10 000-year compliance period of 10 000 years, the Court did not vacate these standards since NAS made no recommendation regarding ground-water standards. Therefore, EPA did not propose changes to the ground-water standards.

The Agency also proposed to update the dose methodology and to revise certain definitions to achieve consistency with the extended compliance period.

Individual-protection standard

The Court's decision centered upon the NAS recommendation regarding the compliance period for the individual-protection standard. To address the Court decision, EPA proposed a compliance limit of 3.5 mSv (350 mrem) CEDE/yr to apply for projected performance between 10 000 and 1 millions years. In addition, EPA proposed retaining the 150 µSv (15 mrem) CEDE/yr standard applicable for the first 10 000 years as established in the 2001 standards.

The Agency believes that the most problematic aspect of extending the compliance period to peak dose is the uncertainty involved in making projections over such long time frames. Regardless of the level of rigor that can be applied to the technical calculation, it is not possible to place the same level of confidence in performance projections over 10 000 years versus 1 million years.

In addressing how to incorporate extremely long-term projections into a regulatory process and have them be sufficiently reliable to serve as a basis for regulatory decisions, EPA considered guidance and precedents from international and domestic sources. The NAS discussed some technical aspects of uncertainty. For example, NAS stated: "uncertainties in waste canister lifetimes might have a more significant effect on assessing performance in the initial 10 000 years than in performance in the range of 100 000 years." [3] On the other hand, NAS recognised that: "the timing of seismic events is unpredictable." [3] Unfortunately, NAS provided no recommendations on how to deal with such uncertainties, but noted: "No analysis of compliance will ever constitute an absolute proof; the objective instead is a reasonable level of confidence in analyses that indicates whether limits established by the standard will be exceeded." [3] For regulatory compliance within 10 000 years, EPA identified several U.S. regulatory programmes as possible precedents, including those for the Waste Isolation Pilot Plant and EPA's underground injection control programme, but for a compliance period extending to 1 million years, there are no precedents in U.S. regulation. In response to the Court decision, therefore, important sources for guidance and models for contemplating regulations at such long times were international programmes grappling with the same issues. In general, international guidance reinforces two points. The first is that uncertainties generally increase with time. For example, the International Atomic Energy Agency [5], the Nuclear Energy Agency [6], and the Swiss National Co-operative for the Disposal of Radioactive Waste [7] have all concluded that the further into the future projections are made, the greater the uncertainty. The second point is that projections at those longer times cannot be viewed with the same level of confidence as shorter-term projections. As exemplified in statements by IAEA [5], NEA [6], and SSI [8] experts indicate that the uncertainties in quantitative performance projections become so large that the results need to be viewed more as qualitative, rather than quantitative, projections.

A number of international scientific and regulatory bodies and programmes suggest natural sources of radioactivity serve as a point of comparison when uncertainties become significant. For example, IAEA has stated that, for time frames extending from about 10 000 to 1 million years, "it may be appropriate to use quantitative and qualitative assessments based on comparisons with natural radioactivity and naturally occurring toxic substances" [9]. The IAEA also suggests that "[i]n very long time frames…uncertainties could become much larger and calculated doses may exceed the dose constraint. Comparison of the doses with doses from naturally occurring radionuclides may provide a useful indication of the significance of such cases [5]. Similarly, NEA stated that a key performance indicator could be "comparison with background radiation levels" for times up to just 100 000 years [6].

The proposed rule describes a dose limit – to apply for the period from 10 000 to 1 million years – that will not cause people living near Yucca Mountain to receive a total dose that is more than the natural background radiation which people receive routinely in other parts of the U.S. In order to assess total exposures and derive a dose limit, it is necessary to establish levels of natural background radiation already experienced in the vicinity of Yucca Mountain. The Agency proposed Amargosa Valley as the point of comparison for this analysis since that is where the RMEI will likely live. Combined with the cosmic and terrestrial exposures estimated by DOE, EPA estimated the total annual natural background radiation in Amargosa Valley to be approximately 3.5 mSv (350 mrem) CEDE/yr.

To make the comparison with total exposures, it is also necessary to consider what total exposures provide a reasonable reference point for limiting releases from Yucca Mountain. As noted above, the goal is to ensure that releases from Yucca Mountain will not cause total exposures of the RMEI to exceed natural background levels with which other populations live routinely. The Agency considered several factors in this selection. First, some incremental exposure will be allowed since the standards cannot be expected to reduce natural background exposures. Thus, the reference point would have to have a higher level of background than does the area near Yucca Mountain. Because of the complications in estimating localised background radiation (due primarily to the radon component), statewide averages, which are less uncertain, were examined. Of the States with sufficient data, 32 have average background radiation levels higher than Nevada. The States' characteristics, such as geographic location and population, were then considered. Colorado was proposed as a State in the western part of the country that best fit the search criteria – fairly well populated and with characteristics reasonably comparable to Nevada (such as radon potential, surface water/coastal features, or size of major cities). According to population data, Colorado ranks 22nd among all states in total population (Nevada is 35th) [10]. Colorado's average annual background radiation is estimated to be about 7 mSv (700 mrem)/yr [11]. Other States have comparable or higher radon potential and higher background levels with which people live routinely (e.g., background levels in North Dakota, South Dakota, and Iowa, for example, are about 8 mSv (789 mrem)/yr, 10 mSv (963 mrem)/yr, and 8 mSv (784 mrem)/yr, respectively), and might also be used for comparison, but their population and geographic characteristics are much different than Nevada's.

Finally, comparing Colorado's estimated average annual background radiation of 7 mSv (700 mrem) CEDE/yr to the estimate for Amargosa Valley, EPA derived an incremental exposure level of 3.5 mSv (350 mrem) CEDE/yr, which was proposed as the dose limit.

The Agency also considered other possible dose limits to apply out to 1 million years. The first option was 1 mSv (100 mrem) CEDE/yr. This level is based upon international guidance to limit all sources of exposure except natural, accidental, and medical. However, in view of the uncertainties in estimating performance in the very far future, EPA concluded that comparisons with natural background radiation provide a reasonable indication of safety out to 1 million years. As McCombie and Chapman have stated in their authoritative reference on radioactive waste disposal: "There is no logical or ethical reason for trying to provide more protection than the population already has from Earth's natural radiation environment, in which it lives and evolves…it must be recognised that man cannot be expected over infinite times to do much better than nature." [12] The other limit considered was 2 mSv (200 mrem) CEDE/yr. It was derived using an approach that incorporated statewide background levels in all the contiguous States in the U.S. However, EPA concluded that it was most appropriate to use site-specific information related to Amargosa Valley (and the RMEI) rather than generic points of reference. For these reasons, the 3.5 mSv (350 mrem) CEDE/yr dose limit, including consideration of natural background radiation in Amargosa Valley, was preferred over the other options considered, and was proposed as the regulatory limit.

We recognised that a standard based on variations in natural background radiation would be higher than previous, non-occupational standards in the U.S. In the 2001 rulemaking, the 150 µSv (15 mrem) CEDE/yr dose limit and the 10 000-year compliance period were justified in part because they were consistent with other EPA policies. However, the circumstances in the proposed Yucca Mountain standards – and, in particular, the nature and degree of uncertainty in projecting performance out to 1 million years – are significantly different from the situations addressed under Superfund or any other existing U.S. regulatory programme. The approach and the dose limit that EPA proposed for the Yucca Mountain standards are consistent with international guidance on the issue of radioactive waste disposal over extremely long times.

Human-intrusion standard

While the Court did not specifically address the human-intrusion standard, the Agency proposed revisions to it to parallel the changes proposed for the individual-protection standard. To do so is consistent with the NAS recommendation that "EPA require that the estimated risk calculated from the assumed intrusion scenario be no greater than the risk limit adopted for the undisturbed-repository case" [3].

The Agency proposed to extend the compliance period from 10 000 to 1 million years and to establish a dose limit of 3.5 mSv (350 mrem) CEDE/yr, which corresponds to the proposed individual-protection dose limit. Other aspects of the human-intrusion standard are unchanged from 2001. The intrusion scenario described in 2001 would still apply because the longer compliance period does not in any way affect the reasoning underlying the selection of this scenario. It remains fully consistent with the NAS conclusion that at Yucca Mountain "there is no scientific basis for estimating the probability of intrusion at far-future times" [3]. Instead, NAS recommended that "the result of the analysis should not be integrated into an assessment of repository performance based on risk, but rather should be considered separately. The purpose of this consequence analysis is to evaluate the resilience of the repository to intrusion" [3].

The intrusion scenario requires consideration of package degradation, premised on the assumption that drillers encountering an intact package would cease drilling and releases would be avoided. We believe that this assumption is equally valid both within and beyond a 10 000-year time frame. In the 2001 standards, DOE was not required to demonstrate compliance with a dose limit if packages did not degrade sufficiently within 10 000 years to permit intrusion (or, in any event, if the consequences of the intrusion were not calculated to occur within 10 000 years). However, the current proposal would require DOE to show compliance with a dose limit regardless of when the consequences of the intrusion occur (within 1 million years). Overall, this scenario continues to represent a reasonable test that "can provide useful insight into the degree to which the ability of a repository to protect public health would be degraded by intrusion" [3].

Dose Methodology

In 1977 and 1979, ICRP published Reports 26 [13] and 30 [14], respectively. These two reports reflected advances in the state of knowledge of radionuclide dosimetry and biological transport of radionuclides in humans that occurred over the 20 years since ICRP's 1957 dose methodology recommendation (ICRP 2) [15]. The new methodology was called the effective dose equivalent (EDE).

The 2001 standards required DOE to calculate annual doses (as CEDE) to demonstrate compliance with the storage, individual-protection, and human-intrusion standards. The Agency proposed to modify that requirement to incorporate updated scientific factors necessary for the calculation, but would not change the underlying methodology. Specifically, EPA proposed to require DOE to calculate the annual CEDE using the radiation- and organ-weighting factors in ICRP Publications 60 [16] and 72 [17], rather than those in ICRP Publications 26 [13] and 30 [14].

These ICRP factors represent the most recent science and dose calculation approaches in the area of radiation protection. The EPA believes that it is reasonable and desirable to conform the standards to the most recent method approved by the U.S. and international radiation-protection community. The Agency also proposed an updating mechanism since repository closure and license termination may be decades or even more than one hundred years into the future. Therefore, EPA would allow DOE to use, with NRC approval, further updated dose calculation factors in the future, but only if those factors have been appropriately reviewed and accepted by the scientific community and issued by independent scientific bodies (such as ICRP and its successor bodies) and incorporated by EPA into its Federal Guidance.

Judging Compliance

Under 40 CFR Part 197, EPA requires DOE to complete a probabilistic performance assessment to demonstrate compliance with the individual-protection standard. The results will be a distribution of projected doses since the analysis contains parameters with a range of values, incorporates uncertainties in the models, and uses various expert-judgment assumptions. In 2001, EPA specified the mean of the distribution as the metric to be used for comparison with the standard. In 2005, EPA proposed to retain the mean as the compliance measure for the first 10 000 years. In the unlikely event that the peak dose is found to occur within the first 10 000 years, the mean would be consistent with the statistical measure used in other applications for geologic disposal, i.e. 40 CFR parts 191 and 194 for the 10 000-year compliance period. However, for the period from 10 000 to 1 million years, the Agency believes that the compliance measure should be examined separately to determine if there is a more appropriate measure.

There are significant uncertainties in predicting when discrete events, such as seismic activity, will occur and the effects of these events. Some scenarios incorporating these uncertainties would represent unlikely behaviour in that they could show extremely poor or extremely good performance. Such low-probability situations should not be ignored in compliance decisions, but they should not be given undue influence in judging compliance. The NAS stated: "The challenge is to define a standard that specifies a high level of protection but that does not rule out an adequately sited and well-designed repository because of highly improbable events."[3]. The Agency concluded that for the longer compliance period, there should be a measure that represents the "central tendency" in the distribution. Therefore, the compliance measure should represent a central measure that is not strongly affected by extreme input and results.

A difficulty with the mean is that when the bases of the calculations are more conservative (or non-conservative), the results suggest that the "most likely" dose is higher (or lower) than if a more realistic approach were taken. Therefore, we believe that a regulatory performance measure should not give undue emphasis to high-end or low-end projections which the mean could do.

On the other hand, the median is less affected by the extremes of the distribution and the attendant uncertainty about how close the mean is to the center of the distribution is removed. In this respect, the median is an attractive alternative to the mean as a measure of central tendency since it is not as strongly influenced by high or low-end outliers. Therefore, EPA proposed to use the median for the post-10 000-year compliance period.

Features, Events and Processes

The overall purpose of the performance assessment is to provide a reasonable test for compliance with the standards. A major part of providing that reasonable test is determining which features, events, and processes (FEPs) are to be included in the performance assessment. Key to this consideration is EPA's goal of setting standards that provide for a reasonable test of the disposal system under a range of conditions that represent the expected case, as well as relatively less likely (but not wholly speculative) scenarios with potentially significant consequences. As a result, it is neither constructive nor necessary for EPA to require DOE to predict or model every conceivable scenario that could occur at Yucca Mountain.

This implies that some FEPs (or series of FEPs) need not be included in the performance assessment because their probability of occurrence is extremely low. As a means of restricting scenarios, in the 2001 standards, the Agency outlined how to screen FEPs. Without such measures, the list of FEPs would be limitless, bounded only by the imagination. The Agency determined that FEPs that could occur with a probability equal to or greater than 1 in 10 000 over a period of 10 000 years, an annual probability of occurrence of 10^{-8}, would be sufficiently likely to occur that they should be included among the FEPs available for selection in any particular scenario. Any FEPs with lower probabilities could be excluded from the performance assessment.

For the 10 000-year to 1-million-year compliance period, we considered how to address this probability cutoff. If, for example, we required consideration of events with a probability of occurrence of 10^{-4} over 1 million years, an approach that has been suggested by some stakeholders, it would equate to an annual probability of 10^{-10}, which encompasses events nearly as remote as the "Big Bang" that created the Universe. No disposal system, and perhaps not even the Earth, would survive the effects of such an event, and, therefore, EPA did not find such FEPs to be useful indicators to distinguish between safe or unsafe performance of the disposal system. In the end, the Agency proposed to retain the screening criterion without change – except as described below. However, certain scenarios merit special considerations at extremely long times (beyond 10 000 years).

The Agency also considered what scenarios should be included in the performance assessment. In formulating our approach to the extended compliance period, we began by reviewing the NAS Report. The NAS concluded that volcanism, seismic activity, and climate change have the potential to significantly modify the properties of the repository and the processes by which radionuclides are transported. The NAS also concluded that the probabilities and consequences of modifications generated by volcanism, seismic activity, and climate change are sufficiently boundable that they should be included (along with an undisturbed scenario) in performance assessments that extend over 1 million years. Thus, EPA proposed to include igneous, seismic, and climate change scenarios and have DOE assess the most likely and significant impacts, with appropriate variability incorporated, on dose projections.

Having identified particular natural FEPs, the Agency considered whether there are FEPs that could significantly affect the engineered barrier system that had not been identified for the 10 000-year compliance period. After reviewing DOE's published assessments and other relevant information, the Agency concluded that general corrosion of the waste packages could be a significant failure mechanism at times in the hundreds of thousands of years [18]. Unlike certain other corrosion processes which may be likelier or faster-acting at earlier times, general corrosion may not be a significant factor within 10 000 years and could potentially be removed from consideration at those times because of its limited consequence. This is a situation that EPA found inappropriate and proposed that DOE must project the effects of general corrosion throughout the compliance period.

Summary of Comments

The Agency received many comments both at public hearings and in writing. In total, there were about 2 550 sets of comments; of those, 2 350 were in mass mailings. We received 3 000 pages of comments and 1 100 pages of attachments from a wide range of individuals and organisations, including members of the U.S. Congress.

- Nevada Congressional members, the Nevada State Governor's office and local governments.
- Tribal governments.
- Other states.
- Nuclear industry groups.
- Environmental and public interest groups.
- DOE.

The comments addressed many aspects of our proposed standards and represent a wide range of views. Some topics were the subject of many comments. They include:

- Post-10 000-year dose limit.
- Groundwater compliance period.
- Mean vs. median.
- Background radiation as an indicator.

Some commenters objected to our approach; others supported it. The Agency is fully considering and will respond to all significant issues raised in comments during the development of the final rule.

Status and Future Steps

The EPA published the proposed amendments to 40 CFR Part 197 in the 22 August 2005 *Federal Register* [19]. A public comment period was open from then until 21 November 2005. Public hearings were held in Amargosa Valley, Nevada; Las Vegas, Nevada; and Washington, DC. Approximately 2 500 comment messages were received.

The Agency is preparing to publish its final amendments. The Agency will considering the comments received and will publish its response-to-comments document and the final versions of its technical support documents when the final amendments are published.

The Yucca Mountain standards and supporting documents may be accessed on the EPA World Wide Web site at http://www.epa.gov/radiation/yucca. There is also a toll-free telephone information line: 1-800-331-9477. The official docket contains all the comments and is accessible at: www.regulations.gov under Docket ID EPA-HQ-OAR-2005-0083.

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Appendix C

SESSION III

RECENT EXPERIENCES IN DEVELOPING A SAFETY CASE

APPLICATION OF SAFETY CASE CONCEPT IN PRACTICE
PRELIMINARY FINDINGS FROM THE NEA INTESC INITIATIVE

J. Andersson
JA Streamflow AB, Sweden

Abstract

During 2006 NEA issued a questionnaire on "INTernational Experiences in Safety Cases" (INTESC). Answers have been received from 15 organisations representing both implementing organisations and regulatory authorities in 9 countries. NEA is currently compiling the answers entailing a detailed review, sorting, and analysis of results to further refine the themes, trends, significant advancements, points of divergence, and key challenges. All responding programmes are preparing extensive safety cases (or preliminary ones) in line with most of the elements of a safety case suggested by the NEA Safety Case brochure. Judging from the regulatory responses, such an ambition level is also required according to regulations. Overall, there are similar approaches and attitudes in different programmes and similar concerns expressed from the participating regulators. Implementers appear to address issues raised by regulators. However, there are some important examples of differences in use or in interpretation and there are some elements of real life safety cases not covered by the brochure. Furthermore, there is substantial overlap between several of the elements listed in the brochure and some further definition of the elements and terminology may be helpful to clarify the actual differences and similarities in safety cases.

Introduction

The nature and purpose of the safety case has recently been described in a Nuclear Energy Agency (NEA) publication [1]. According to this publication, a safety case is "the synthesis of evidence, analyses and arguments that quantify and substantiate a claim that the repository will be safe after closure and beyond the time when active control of the facility can be relied on". The safety case becomes more comprehensive and rigorous as a programme progresses, and is a key input to decision-making at several steps in the repository planning and implementation process. A key function of the safety case is to provide a platform for informed discussion whereby interested parties can assess their own levels of confidence in a project, determine any concerns they may have about the project at a given planning and development stage, and identify the issues that may be relevant to safety and on which further work may be required.

Recently, there has been notable convergence in documents published by national and international organisations in the understanding and development of long-term safety cases for geological disposal. The final draft of the IAEA safety requirements guide [2], the NEA brochure on the safety case [1], the presentations at the Stockholm conference of 2003 [3], and several recently released safety reports from different countries all show a significant assimilation of the principles by national organisations that were set forth in the NEA Confidence Report [4] and subsequent documents, such as reports of the NEA project on Integrated Performance Assessments of Deep Repositories (IPAG) [5].

At the 38th RWMC meeting members approved in principle to set up an exercise on "INTernational Experiences in Safety Cases" – namely the INTESC based on the results of a questionnaire.

The exercise has the following primary aims:

- to analyse existing safety cases and elements or components of safety cases that are under development and to identify key concepts, including points of consensus and divergence;
- to provide a clear overview of the progress that has been made in the last decade;
- to provide a clear overview of regulatory expectations on the future safety cases;
- to report the state of the art to the IGSC and as an input to the forthcoming international symposium of January 2007;
- to report on practical experiences on safety cases for geological repositories and the lessons learnt from current practices.

A questionnaire addressing these aims was developed and approved under the auspices of the NEA Integration Group for the Safety Case (IGSC). Answers have been received from 15 organisations representing both implementing organisations and regulatory authorities in 9 countries. NEA is currently compiling the answers entailing a detailed review, sorting, and analysis of results to further refine the themes, trends, significant advancements, points of divergence, and key challenges. Using this compilation a synthesis report will be produced to describe current practices in the implementation of safety cases and to be used as input to NEA IGCS group's exploration of future directions for work. This paper provides some preliminary observations from the ongoing compilation and assessment work.

Purpose and context

There is a wide range of programme development represented in the questionnaire answers. Most of the reported Safety Cases have been prepared for an actual license application or are being prepared for a coming license application. However, many apply to earlier stages of development and concern more generic feasibility or are prepared to guide further R&D. Furthermore, several respondents refer to uncompleted – or even planned – Safety Cases. Generally, the progress of the cited Safety Case reflects the status of the national programme, with more generic examples from programmes at the stage of generic feasibility studies, and more specific ones for programmes in the process of selecting a site or sites for characterisation from the surface or to license an underground repository.

The responses mainly concern high-level waste (HLW) and spent nuclear fuel projects. However, there are also answers from intermediate-level waste (ILW) repository projects.

Most regulatory answers concern existing regulations on the Safety Case, but experiences from previous reviews are also addressed. With few exceptions, the regulatory answers do not address whether the existing safety cases presented by the related implementing bodies fulfil the regulatory requirements.

Commonly used elements of the safety case - areas of general agreement

Recognising that format and content should be adapted to the decision-context of each safety case, the NEA Safety Case Brochure [1] describes the elements that may contribute to the safety case as follows:

- *The safety strategy:* The safety strategy is the high-level approach adopted for achieving safe disposal, and includes an overall management strategy, a siting and design strategy and an assessment strategy. Management strategies should accord with good management and engineering principles and practice. The siting and design strategy is generally based on principles that favour robustness and minimise uncertainty including the use of the multi-barrier concept. The assessment strategy must ensure that safety assessments capture, describe and analyse uncertainties that are relevant to safety, and investigate their effects.
- *The assessment basis*: The assessment basis is the collection of information and analysis tools supporting the safety assessment. This includes an overall description of the disposal system (that consists of the chosen repository and its geological setting), the scientific and technical data and understanding relevant to the assessment of system safety, and the assessment methods, models, computer codes and databases for analysing system performance.
- *Evidence, analyses and arguments:* Most national regulations give safety criteria in terms of dose and/or risk, and the evaluation of these indicators, using either mathematical analyses or more qualitative arguments, for a range of evolution scenarios for the disposal system, appears prominently in all safety cases that are intended for regulatory review. Complementary types of evidence and arguments in support of a case for safety include general evidence for the strength of geological disposal as a waste management option, evidence for the intrinsic quality of the site and design, and safety indicators complementary to dose and risk.
- *Synthesis:* A safety case may include a statement of confidence on the part of the author of the safety case based on the analyses and arguments developed and the evidence gathered. To this end, a synthesis of the available evidence, arguments and analyses is made to highlight the grounds on which the author has come to a judgement that planning and development of the disposal system should continue.

The subsequent discussions describe the results of the INTESC questionnaire in terms of these elements of the safety case.

Safety strategy

Judging from answers all national programmes aim at management strategies that accord with good management and engineering principles and practice e.g. quality plan, how to adapt to stakeholders requirements, allocation of resources and co-ordination activities. Most programmes apply a stepwise approach to decisions. All programmes declare a flexible approach to design and focus of the safety case work. Most programmes are subject to formal quality assurance (QA), with safety assessment QA plans being part of the overall Quality Management System of the organisation. Experts are selected on the basis of references and expert decisions are documented. Effective integration of information and knowledge coming from the different fields of the programme is, for example, ensured by a high-level overall integration team. Key safety issues govern program priorities. All programmes make provisions for storing information in some type of document management systems.

Uncertainty assessment is a key component of the assessment strategy in most cases. In most assessments deterministic and probabilistic calculations are seen as complementary and both approaches are adopted. Conservative assumptions are acceptable to handle situations where knowledge is lacking or phenomenon is poorly understood. Regulators generally accept stylised approaches for assessing certain aspects of scenario analysis, such as for assessing future human actions and for assessing the biosphere evolution. However, the extent of stylising is still a matter for interpretation. Many programmes consider "what-if" cases, usually postulating loss of barrier function(s). However, their rationale, as well as the view whether such scenarios lie outside the risk contribution slightly varies between respondents. Several examples of handling diverse opinions are applied.

Assessment basis

There is a wide agreement that the describing the system concept is a key part of the safety case. There is a general agreement that the biosphere should not fulfil any safety functions. However, its properties influence especially how groundwater contamination is distributed in the human environment and this needs to be assessed.

According to the Safety Case Brochure the presentation of scientific data and understanding in a safety case should highlight evidence that the information base is consistent, well founded and adequate for the purposes of safety assessment. Also the assessment methods, models, computer codes and databases must also be clearly and logically presented. All respondents generally support these ambition levels and there are several examples of specific actions for meeting these goals.

Evidence, analyses and arguments

In evaluating performance/safety indicators several actions are, indeed, taken for checking reliability or plausibility of safety cases including: validation against large scale experiments or field data, various QA procedures, the iterative approach in which the safety case is progressively refined and reviewed, comparison with simplified analytic approaches, peer review and various international exercises. Some of the cited assessments show doses and/or risks at or above acceptance limits, in altered evolution scenarios or other cases of low likelihood – or at very late times. However, in light of the uncertainties, these cases are not seen to violate eventual compliance with regulation. Safety and performance indicators in addition to dose and risk are also used, mainly for illustrative purposes and the selection reflect the measures that have been discussed in various international fora.

There is a general aim to demonstrate that the system can be implemented with existing technology and according to the responding implementers this has also generally been achieved. Defects in manufacturing or implementation are taken into account by a varying degree, largely dependent on the repository concept and on the importance of such defects in terms of safety. Generally, the strategy to deal with remaining uncertainties is to assess them and devise plans for sufficient resolution through ongoing R&D, site investigation and repository design projects.

Complementary evidence and lines of argument are also presented. The benefit of a systematic and documented identification of safety functions and criteria for the safety functions is also mentioned.

Synthesis

Some assessments already conclude that there is adequate confidence in the possibility of achieving a safe repository to justify a positive decision to proceed to the next stage of planning or implementation. Other assessment intend to when the safety case will be presented as part of an application. Furthermore, all assessments contain at least preliminary conclusions regarding safety.

Presenting the safety case

The safety case is mainly documented for a technical audience, primarily for review by the regulator. It is usually presented in a main safety report supported by several main references and a number of lower level reports. It may also be noted that most regulators have issued, or plan to issue, documents on how they will review a safety case.

In applicable cases a summary of the Safety Case is, or will be, presented in the Environmental Impact Assessment, as well as shorter summaries and brochures are prepared for wider audiences. Only some respondents appear to give an important role to other types of media in addition to printed documents in presenting the safety case to different audiences (e.g. computer graphics, videos).

Elements not used by all or given different interpretation

What elements are developing and gaining in use?

Several safety case elements listed in the brochure are developing and are gaining in use:

- Application of data clearance procedures.
- Development of actions for very long-term preservation of information.
- Development of a siting and design strategy in which final adjustment of layout will be done according to the specific characteristics of the rock that is found in advancing the construction steps.
- Using safety function indicators as a tool for scenario selection.

Where are there variations in interpretation?

There are several areas in the siting and design strategy of varying interpretation:

- While several barriers are required, strict application of the multi-barrier-principle is not required in all programmes.
- Not all countries require the use of "best available technology" (BAT). The concept is also subject to interpretation, although it appears generally accepted that due consideration should be given to economical and other societal factors i.e. taking into account what is reasonably achievable in making design decisions.
- Post closure monitoring is planned by most organisations but this is not necessarily seen as key component of the safety case.
- Most allow for retrievability but only few programmes take actions on revising disposal concepts to dramatically facilitate this.

There are also varying interpretations of some elements of the assessment strategy:

- Most regulators, but not all, accept a separate treatment of Future Human Action scenarios. Many, but not all, assessments consider the risk to the intruder.
- Most implementers apply some alternative conceptual models of site or repository behavior, but the depth of consideration of alternative models is a matter of discussion between implementer and regulator.

Some respondents appear to focus on the radionuclide retention aspects of the safety functions. Others put most of the emphasis on the containment (isolating) functions and derive safety functions related to the ability of the system to provide containment.

Transparency as well as traceabilty are of key importance from a regulatory perspective, but few respondents addressed how to handle conflicts between transparency and traceability.

Which seem to be falling out of favour or not yet widely used?

Only one respondent directly addressed the general strength of geological disposal in the safety assessment document. Possibly this is an issue to be (or have been) addressed at the national policy level; but of course, the analyses and safety case must support a conclusion on the safety of geological disposal as an overall waste management strategy.

Judging from responses there is little distinction between arguments presented for safety and confidence and "complimentary arguments". This division is probably not really helpful. Few respondents use the term "reserve FEP", even if the generally conservative approach in Safety Assessment is indeed brought forward.

Elements of safety cases not reflected or beyond the ones in the brochure

There are elements in real-life safety cases that are not reflected in, or go beyond, the ones in the brochure. The possibly two most important examples of this are:

- Preparation of a geosynthesis, i.e. assessing geoscience information from a variety of perspectives such as structural geology, hydrogeology, and geochemistry and synthesising this data into an integrated geosphere model that is consistent with the knowledge and history of the site.
- Account of the construction and operational period. Some respondents systematically address the thermal, mechanical, hydraulic and chemical processes/alterations for this stage, using the same methodology as for subsequent, post-closure stages, whereas other still develop their approach – or even question whether it is important for post closure safety.

Conclusions

All responding programmes are preparing extensive safety cases (or preliminary ones) in line with most of the elements of a safety case suggested by the NEA Safety Case brochure. In fact, judging from the regulatory responses, such an ambition level is also required according to regulations. Overall, there are similar approaches and attitudes in different programmes and similar concerns expressed from the participating regulators. Implementers appear to address issues raised by regulators. However, there are some important examples of differences in use or in interpretation and there are some elements of real life safety cases not covered by the brochure. Furthermore, there is substantial overlap between several of the elements listed in the brochure and some further definition of the elements and terminology may be helpful to clarify the actual differences and similarities in safety cases.

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BÁTAAPÁTI REPOSITORY – EVOLUTION OF THE SAFETY CASE

M. Goldsworthy
Golder Associates, Germany

P. Molnar, G. Danko
Golder Associates, Hungary

Z. Nagy, F. Frigyesi
PURAM, Hungary

Introduction

As expressed in the NEA document [1] a safety case for a repository for radioactive materials is currently considered to cover many aspects. Some of these aspects are what would be considered management issues, such as establishing strategy, rather than e.g. demonstration of compliance of the performance of the repository system with specific technical criteria. It can reasonably be presumed that these less obviously technical aspects are emphasised with the objective of ensuring a consistently safety-oriented context within which decisions relevant to safety will be made. It is nevertheless the case that, ultimately, the post-closure safety of the repository will be judged using technical criteria, with one standard check of satisfactory performance being the radiological dose received by the person most exposed to any radionuclides released from the repository system.

One of the many challenges in a repository programme is ensuring at any stage that the appropriate levels of organisation and responsibility have been established, and that there is sufficient information available to support the decisions which have to be made. As noted already, criteria will be applied which consider the post-closure performance of the repository system. As this generally depends strongly on the underground conditions around the repository site a particular difficulty arises in this context. Because of the formal requirements and the expenditure involved, undertaking specific investigations of these conditions implies a high degree of stakeholder commitment to the project and even to a particular site. For this reason it can be expected that the complete information required for a reliable evaluation of the post-closure safety will only be available late in the programme.

As a real example of a repository programme the history to date of the evolution of the safety case for the new L/ILW repository in Hungary is described in the paper. Because of the importance of the information about the underground conditions at the site particular emphasis is given to this aspect.

Early stages

The Hungarian programme for the development of a new repository for L/ILW has its roots in the development there of the use of nuclear power. There is an existing repository for institutional waste and sealed sources at Püspökszilagy, but this was not considered to be adequate for the waste from the

4 WWER 440 reactors at the Paks nuclear power plant (PNPP). These reactors were connected to the grid during the 1980s.

An initial screening process was carried out for PNPP by Hungarian FTV Consulting Engineering to identify a site for a new repository. This considered geological and technical criteria, and led to identification of a candidate near-surface site. In the political climate at the time this project became one focus of opposition to the government and it was rejected in a local referendum in January 1990.

Site selection and initial development of the safety case

In 1992 the Hungarian Atomic Energy Commission initiated a National Project for the solution of the issue of low and intermediate level radioactive waste disposal. The search for a repository site was resumed in 1993-1994, using a more open process that considered societal, as well as geological and technical criteria. Screening by the Geological Institute of Hungary and many other institutions was started with a nationwide approach considering both negative and positive attributes. Amongst other things, zones close to national borders, mineral deposits, water resources, karstic systems, military objects, potential landslide areas and recreation areas were excluded in the screening process. Near surface and deep geological repositories (maximum depth 300m) were both considered as a possible solution. Preliminary positive screening (e.g. for upland areas with homogeneous low permeability) revealed that the only area for both types (near-surface or underground) of disposal system was Mezőföld (in the south-western part of Central Hungary) and an area to the south. This was followed by further screening concentrated on the approx. 5 000 km^2 area in Mezőföld. This screening resulted in numerous prospective objects. However, after checking public acceptance only 12 near-surface and 18 underground objects remained. Four prospective areas, three for near surface repositories (Németkér, Udvari and Diósberény) and one for underground disposal (Bátaapáti-Üveghuta), were covered by field reconnaissance. Three boreholes were drilled in 1996, two in potential near surface (loess) areas (Udvari and Diósberény) and one in an area of granitic rock for a potential underground repository (Üveghuta) [2].

In parallel with the site selection process there was also significant development in the legal and administrative aspects of the repository programme. In 1996 the Parliament of the Hungarian Republic enacted the law on atomic energy, which states that the performance of tasks related to radioactive waste management and decommissioning of nuclear facilities is to be accomplished by an organisation designated by the Government. The Public Agency for Radioactive Waste Management (PURAM), the organisation which was established for the purpose, was incorporated in 1998.

Two Hungarian regulations relevant to the repository programme were published in this period. Together these regulations established important parts of the safety strategy - for siting and design, and for assessment.

In 1997 a regulation was issued covering geological requirements for siting of disposal facilities for radioactive waste [3]. The requirements have been summarised [4] as follows:

- The long-term behaviour of the geological environment and the elements of the engineered barrier system can be known and modelled appropriately
- The geological environment is stable in the timeframe necessary for the safe disposal;
- The geological environment and the engineered barriers have isolation properties within which:
  - The hydrogeological system provides for the delayed travel time of the radionuclides to the surface and for the dilution of their concentrations.

  - The geological environment and the engineered barriers constrain the movement of the radionuclides by their retention, retardation and sorption capacity.
  - The geological environment provides safety against the detrimental effects of surface processes and against human intrusion.

- The geological environment and the elements of the engineered barrier system do not reduce the other's effectiveness, but enhance it.

The evaluation of the suitability of a site taking account on these requirements is not to be based on any specific limiting values but rather on the considered opinions of specialists with appropriate expertise.

The aim of the subsequent investigation was to check whether the site fulfils these requirements.

The 1998 regulation [5] (now superseded) defined radiation protection requirements (0,25 mSv effective dose equivalent limit) for the final disposal of radioactive waste.

In 1997 preliminary safety reports were prepared based on concepts for a deep geological repository in the granitic rock and for a near-surface repository in one of the loess areas. For a probabilistic analysis of the post-closure performance of the repository system in granitic rock the information from the existing borehole was taken as a basis. This showed a fractured granitic rock mass with a downward hydraulic gradient indicating infiltration from the surface. Further data needed to characterise the transport phenomena in the fractures was derived from international experience of such sites. Comparison of the results of the safety analyses for the two repository concepts and locations favoured the underground repository.

Also in 1997 a geological evaluation was made of five areas in the neighbourhood of the existing borehole in the granitic area. A site was selected for further study with the objective of confirming its suitability in relation to the geological requirements [3].

Site confirmation

As part of the assessment strategy an external peer review was undertaken during the confirmatory process. The WATRP report [6] was issued in 1999. The summary of the findings of the review team was that the programme was proceeding satisfactorily, although some suggestions were made for improving certain aspects:

- Greater flexibility should be allowed in repository design with emphasis on total system safety based on a combination of engineered and natural barriers.
- Clarification should be made of the design concept and the kinds of engineered barriers to be included in the design.
- The safety assessments that were provided to the team, based on limited early geologic investigations, should be updated. There is a need for an integrated safety assessment using the currently available site and conceptual design information, and including a broader spectrum of scenarios. This integrated safety assessment should form the basis for continued site characterisation, and preferably be prepared, at least in part, before presenting the case to Parliament.
- The safety assessments to date have focused on long term performance. As the design concept matures, there is a need to consider potential radiation exposures of workers and the public, as well as conventional mine safety, during repository operation.

In 2000 a short geological summary was prepared, providing a synthesis of the then current understanding. On this basis a safety analysis was performed. This was based on a previous safety analysis undertaken as part of an international cooperation. In contrast with the preliminary analysis made in 1997 the conceptual model assumed that the granitic rock would be able to act as diffusive barrier zone around the repository. The results were that the radiological risk to the public is negligible for the post-closure phase for both normal and altered evolution scenarios [7].

The work carried out in the further investigations up to 2003 was summarised in an annual report from the Geological Institute of Hungary in 2004 and it was also published [8]. These papers documented the scientific and technical understanding of the geological and hydrogeological conditions up to that time.

The report was the basis of an evaluation of the geological suitability of the site area made by the responsible public authority the Regional Office of the Hungarian Geological Survey The formal confirmation of the geological suitability of the site in accordance with the regulatory requirements [3] was issued in 2004.

One challenge for the geological evaluation of the rock conditions in the site area was that little of the rock was exposed. The area selected for the study was covered with a thick loess layer and exposures of the granitic rock were only found in some streams. This meant that the 6 deep boreholes made in the site area in 2000 were particularly important for the definition of the conceptual model of the underground conditions. One new observation was that there appeared to be sharp changes in the profiles of hydraulic head measured in the boreholes.

A further 8 boreholes were drilled (2 of them outside the site area, in outflow zones) by 2004 as part of the next step, the characterisation programme, and the information from these was combined with that from the previous investigations in the conceptual models used for new safety analyses carried out in that year. By that time it was apparent that there were distinct differences between the northern and southern parts of the site and that there were hydraulically significant features of considerable lateral extent. The data indicated a slow downward flow of water in the fresh rock, but with a higher rate of flow in the northern part. Zones of the site had been identified which have good hydraulic connectivity over considerable distances, and some features were known to act as barriers to flow, resulting in sudden changes of hydraulic head over small distances.

In the safety analyses made in 2004 different approaches were taken for the geosphere. For the probabilistic safety analyses the assumed flow paths were divided into two main parts (a slow path in the background fractures and a fast path in highly conductive features). The slow path sets of fractures were simulated on the basis of the hydraulic and physical measurements made at the site. One of the objectives of these analyses was to evaluate the performance differences between potential repository locations in the north, centre and south of the site area, but with the concept for the repository system essentially unchanged from that investigated in 1997.

During this period a further government decree was issued [9] which defined certain issues for the interim storage and final disposal of radioactive wastes. Appendix 4 to the decree contains the requirements for final disposal and Appendix 5 deals with the safety evaluation of the final disposal.

Site characterisation and extension of the safety case

The ongoing site characterisation work included 3 further deep boreholes, which were drilled in the northern part of the site to improve the investigation coverage there, and 2 additional deep boreholes to investigate the planned path of the access tunnels. The principal work is being carried out in the two inclined tunnels. At the time of writing both tunnels are about 1 000 m long and have reached to about halfway between the portal and the designated site area. It is anticipated that they will attain their full length during 2008.

One of the products of the site characterisation work is an improved understanding of the regional distribution of the zones of granitic rock described respectively as monzonite and as monzogranite. The monzonite, which is found in the northern part of the site area, has been shown to have a high hydraulic connectivity over distances of hundreds of metres. In the tunnels strike-slip fault zones have been encountered. The massive fault gouge in these zones contains large amounts of clay minerals. These zones significantly influence the water flow and pressure distributions. It is thought that zones such as these probably explain the sudden changes of hydraulic head and the slower groundwater movement which are observed in the monzogranite zone. As such fault zones are also associated with increased fracturing of the rock mass their presence also implies worse stability of the underground openings required for a repository.

In parallel with the investigations and evaluations directed towards improved characterisation a number of actions were taken in the programme with the objective of improving the assessment basis of the safety case. One of these was the incorporation of an explicit quality assurance system in the investigation and evaluation processes, and the appointment of an external organisation to implement and apply the system. An expert external reviewer has also been engaged to ensure that the investigations in the tunnels are being carried out in accordance with the state of the art for this kind of work. With the objective of ensuring that the data so gathered are used in a consistent way work has also been carried out on establishing a relational database for the project information.

Current situation – late 2006

In preparation for making an application in 2007 for an implementation licence a design team, with international expert members, is reviewing the repository concept and has already made recommendations for a conceptual design. After its approval, this will be used as the basis of the detailed design to be submitted with the licence application. One recommendation, based on the current information on the underground conditions, as revealed by the investigations and tunnelling, is that the repository should be placed in the monzogranite area.

For the safety report, which is also required to be submitted with the licence application, it is planned to incorporate in the performance analysis the most recent repository concept and the improved knowledge on the location of natural barriers and of their significance for the hydraulic conditions around the repository.

Discussion

It is almost inevitable that the site-specific information which is available initially for a deep geological repository will be extremely limited. This was certainly the case in Hungary and this deficiency presents a considerable challenge in the context of establishing a safety case for use in making the important decisions on site selection and on finance which are necessary in the early part of such a repository programme.

It is clear that simply relying on precedent experience cannot always be relied upon as being sufficient. One obvious problem with that approach is the complete absence of direct evidence that the long-term performance of any repository is adequate. We might neglect this issue, on the basis that there are many competent studies indicating that it will be, but the case of Bátaapáti illustrates another problem too. That is that an initial conceptual model, especially if linked implicitly to one of the international standard conceptual models; crystalline, clay, salt etc., may be shown later to have been inadequate. Or at least inadequate in the sense of not including some features of the real underground conditions which are important for the performance of the repository system.

One simple response to this rather likely situation is that in all stages of the project the possibility of divergences from the initial conceptual models must be stated explicitly, even at the risk of stakeholders having a lower level of confidence. The deliberate selection of a safe model could counteract this risk, as it would enable the statement to be made that the real conditions, if actually different, will certainly be safer or less costly.

The second conclusion relates to the risk that the implications of a revised conceptual model could come to be regarded as inconvenient in the context of the project objectives. It is perhaps for this reason that the safety case covers many aspects of a repository programme, as was observed in the introduction.

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AN ITERATIVE APPROACH TO ACHIEVING SAFETY: APPLICATION IN THE DOSSIER 2005 ARGILE

B. Cahen, S. Voinis, L. Griffault
National radioactive waste management agency, France

Abstract

The "Dossier 2005 Argile" presents the feasibility assessment – with regard to the technical capacity to accommodate all wastes, to reversibility, and to safety – of radioactive waste disposal in a clay formation studied at the Meuse/Haute-Marne URL. It was built upon an iterative approach between site characterisation, design, modelling, and safety analysis, in which two principles always guided the elaboration of the safety case:

- Robustness – repository components must maintain their functionality given reasonable solicitations, taking into account uncertainties on the nature and level of these solicitations.
- Demonstrability – the safety must be verified without requiring complex demonstrations, but based on multiple lines of evidence/argument (numerical simulation, qualitative arguments such as use of natural analogues, experiments and technological demonstrators).

In that respect, key elements in the development of a coherent post-closure safety case are: (i) the functional analysis to determine the safety functions and requirements and the related technical architecture and design based on current industrial experience; (ii) the Phenomenological Analysis of Repository Situations providing a good scientific understanding based on surface and underground laboratory experiments, (iii) the qualitative safety analysis managing uncertainties and the quantitative assessment of scenarios including sensitivity analysis.

The aim of this article is to present the methods that Andra implemented in the context of Dossier 2005 Argile.

Introduction

The December 30, 1991 French Waste Act entrusted Andra, the French national agency for radioactive waste management, with the task of assessing the feasibility of deep geological disposal of High Level Long-lived waste (HLLLW). The emphasis placed on the demonstration of safety [1] was gradually combined with considerations of prudent repository management [2,3]. As a result, two guiding principles – Long term safety and Retrievability – are fundamental requirements inherent in Andra's repository design concept. They are intended to protect the rights of future generations, by providing them with a viable solution, without restricting their control over the waste management process. Of course, other concerns, such as the safety of workers and the protection of the public and the environment during the operation of the facility, are also essential in the design of the facility.

The Basic Safety Rule RFS III.2.f of June 1991 [1], issued by the French nuclear safety authority, provides a framework for the studies to be conducted as such: the protection of man and the environment are to be demonstrated; studies should show the ability to limit potential consequences to a level as low as reasonably possible; the concept should include a multiple barrier system, and rely on passive repository evolution without institutional control beyond a given timeframe (500 years).

The "Dossier 2005 Argile" [4,5,6 and 7] presents the feasibility assessment – with regard to the technical capacity to accommodate all wastes, to reversibility, and to safety – of radioactive waste disposal in a clay formation studied at the Meuse/Haute-Marne URL (eastern France). It was built upon an iterative approach between site characterisation, design, modelling, and safety analysis, in which two principles always guided the elaboration of the safety case:

- Robustness – repository components must meet safety requirements and maintain their functionality given reasonable solicitations, taking into account uncertainties on the nature and level of these solicitations.
- Demonstrability – the safety must be verified without requiring complex demonstrations, but based on multiple lines of evidence/argument (numerical simulation, qualitative arguments such as use of natural analogues, experiments and technological demonstrators).

Although the safety approach of the Dossier 2005 goes back to the concepts and the general spirit of a "conventional" nuclear installation safety approach, it differs from such an approach by a few general aspects:

- The necessity of approaching in a coordinated way the different life phases of the repository (i.e. operation, and post-closure).
- The taking into account of timescales which extend beyond human experience.
- The strong relationship between technical design, scientific knowledge acquisition and safety assessments.
- The key importance given to the notion of uncertainties management in particular, for the post-closure phase.

These peculiarities result as much from the studied object's specificity (the repository in a deep geological formation) as from the question raised (that of feasibility). It requires calling on many disciplines (mining and nuclear engineering, earth sciences, material sciences, safety) and implementing specific methods at the interface between those disciplines.

These principles and objectives need to be taken into account at the core of the engineering and scientific studies. It requires the use of specific management tools, since a variety scientific and technical domains are covered by the studies. In this context, the integration of the scientific knowledge and the definition of a clear safety approach are key elements in the development of a coherent safety case.

Key elements used to assess feasibility of a geological repository in clay

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

a. Basic inputs such as the inventory model of the waste and the geological site.

b. Safety functions and requirement management (operation and post-closure phases).

c. Technical solutions based on industrial experience.

d. Management and monitoring, to allow retrievability (reversibility).

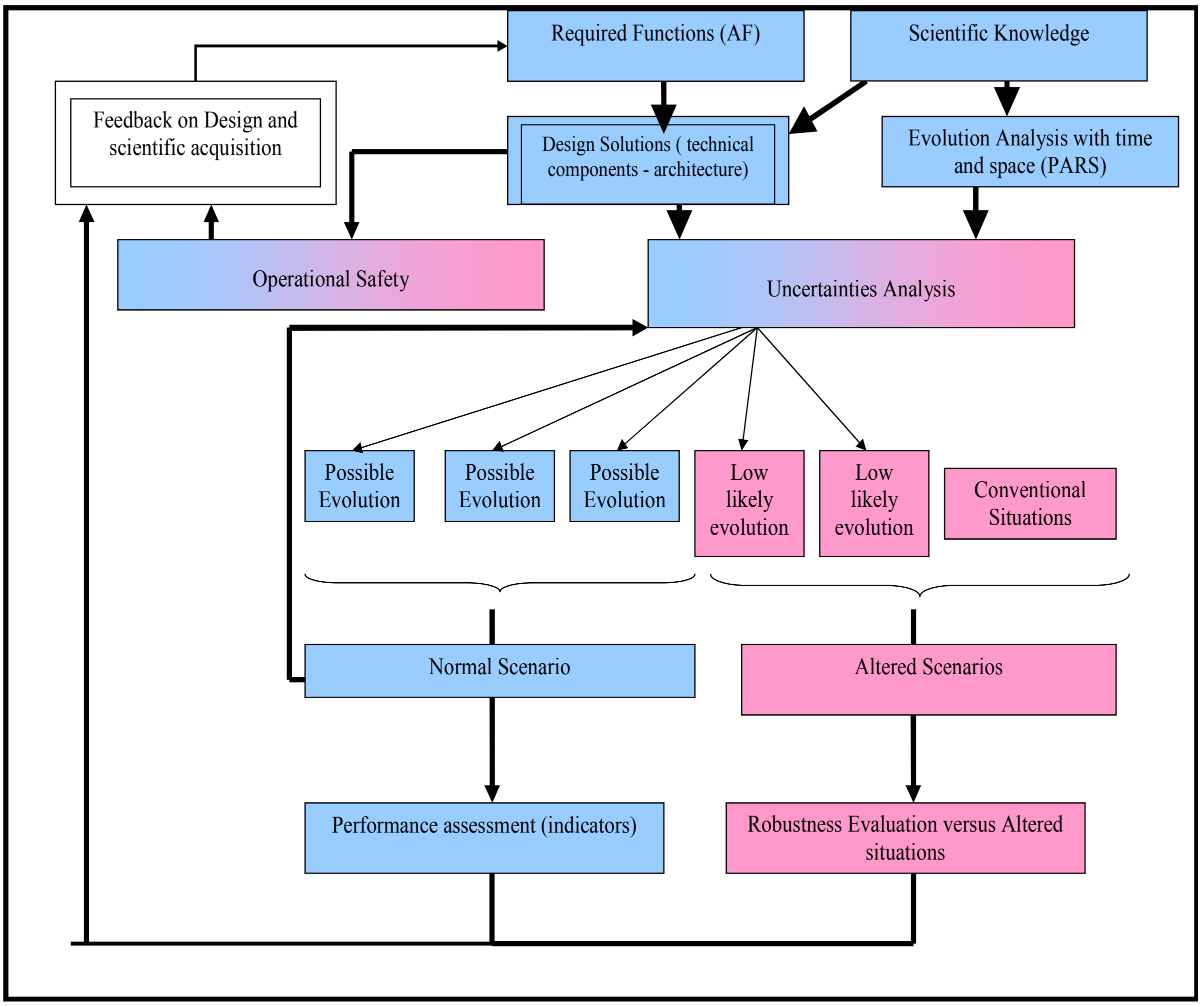
e. Phenomenological Analysis of Repository Situations (PARS) and detailed, coupled process modelling.

f. Qualitative Safety Assessment, namely QSA (Fr: AQS).

g. Simulation platform namely "ALLIANCES" and quantitative assessment results.

The process, thus summarised, is in fact highly iterative, with feedback loops between the various processes as shown in Figure 1. In that context, in view of providing sound feedback to design, research and development and determine residual uncertainties, the following tools have been carried out : the functional analysis (FA) to determine the safety functions and associated requirements – what do we want? the Phenomenological Analysis of Repository Situations (PARS) providing a good scientific understanding based on scientific studies from surface and underground laboratory – what do we get? the qualitative safety analysis (QSA) managing uncertainties and the quantitative assessment [safety and performance indicators] including sensitivity analysis. What is the impact of a given uncertainty (or set of uncertainty factors) on the robustness of the system? And eventually: does the concept meet the safety/acceptability criteria?

Figure 1. **An iterative design approach**



The functional analysis – A systematic method towards technical solutions

With respect to international guidance regarding the main elements of a safety case [8], Andra applied the notion of "multi-safety functions" (i.e. a system of controlling the safety of the repository by assigning safety functions) as a complement to the so-called "multi-barrier" approach. In many ways, the "multi-function" approach is a generalisation of the "multi-barrier" concept relying on the geological layer (host rock), engineered components and waste containers and packages. The approach allows safety to rely on multiple functions performed by various components of the disposal system. Each function is characterised by: a performance level, the period during which the function has to be available and the component(s) (one or more) that have to fulfil the function. This approach acknowledges the fact that the components of a repository may not act as traditional "barriers" once the repository is closed, as total containment may not be guaranteed in the long run. Safety functions give access to a finer definition of the role of each component, making it possible to assess the contribution of each of them to the overall safety performance. It may allow us to identify features that are important for the global safety of the repository, even though they may not relate to a containment capacity.

The fundamental objective of the repository with respect to safety in the basic safety rules RFSIII.2.f consists of "protecting the human being and the environment against hazards associated with the dissemination of radioactive substances" in the short and long term. This objective is formally restated in the functional form 'to protect humans and the environment from the dispersal of radioactive nuclides" and is considered as the main safety function for the post closure phase.

The derivation of the main safety functions, starting from general ones to more detailed ones, is then guided by a systematic methodology classically utilised in other industrial contexts such as aeronautics, and space industries. One of the difficulties of performing such an analysis is to make sure that the set of safety functions that is finally obtained is "comprehensive", meaning that all functions that are relevant and may guide the design of the repository are clearly identified. Since a functional analysis is the expression of a certain state-of-the-art knowledge, it is expected that some functions may be overlooked at first go and added later on. But, at any given time, the functions should mirror the reflections of the implementer. What was used for Dossier 2005 was a method of "flux management", which guided the breakdown structure of the three main safety functions while taking into account water and radionuclide fluxes. Indeed, in the "Dossier 2005 Argile", the flux of radionuclides through the repository was the most important one on the long term, although the flux of water may prove to be important also, even though only small fluxes are expected. In addition, the flux of mechanical constraints inside the repository may need to be considered, as the host rock may be damaged by it.

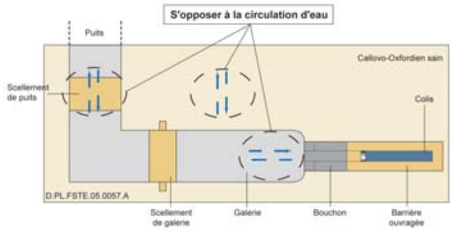
With respect to this method, the fundamental safety function "protecting the human being and the environment against hazards associated with the dissemination of radioactive substances" can be declined into three high-level safety functions, that are at the core of the long-term safety assessment: (i) prevent water circulation in the repository (ii) limit the release of radionuclides and immobilise them inside the repository, and (iii) delay and reduce the migration of radionuclides toward the environment (Figure 2).

Figure 2. **High-level safety functions and components**

Safety functions over time

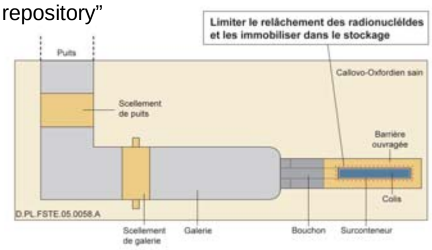
Opposing the circulation of water

- Control the water flow
- Reflects the requirement for sealing resaturation



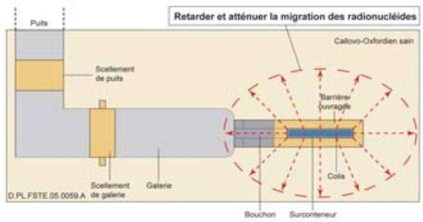
"Limiting the release of radioactive nuclides and immobilising them within the repository"

- Control the thermal flow (C-type waste and SF) During the initial thermal phase
- Reflects the requirement to protect the waste packages from water and place them under favourable physicochemical conditions (pH…).



Delaying and reducing the migration of radioactive nuclides

- Control the radionuclide flow
- Act after radionuclide have been released in the geological medium
- Reflects the transport properties, time for diffusive transport process



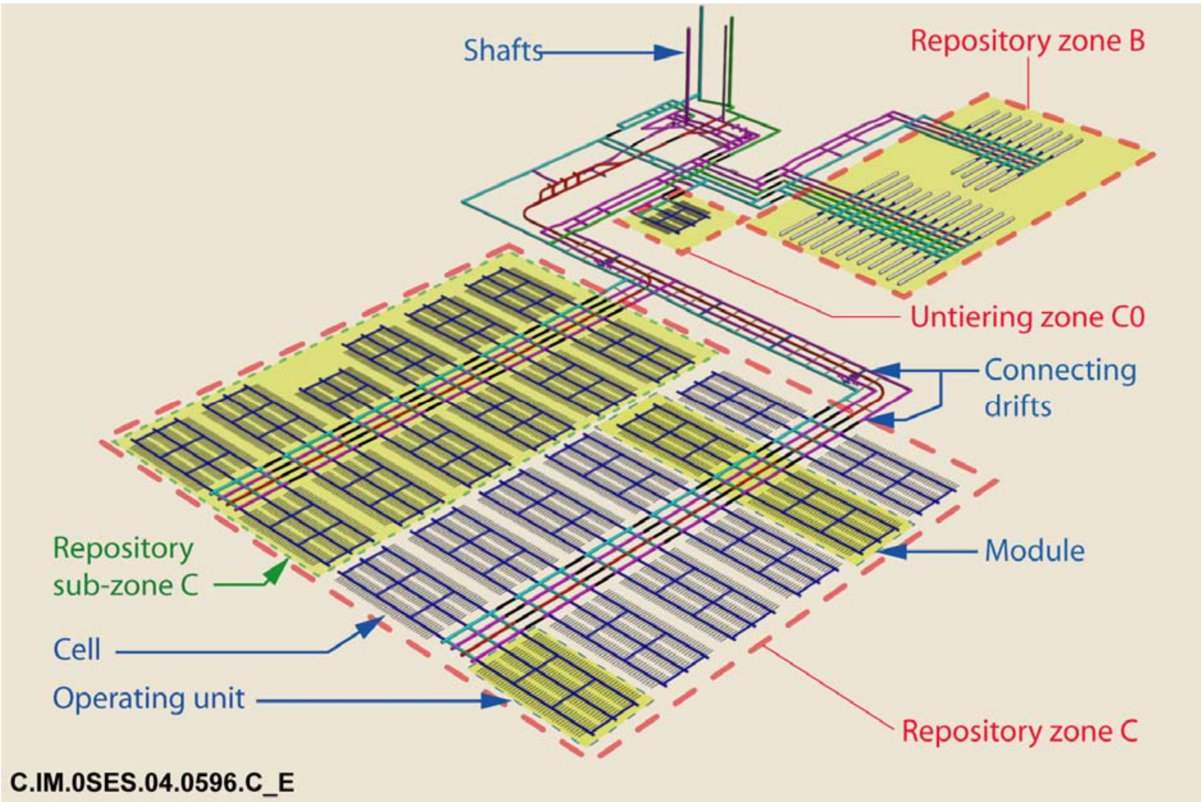
To ensure that safety considerations govern repository design, as well as construction and operating procedures, the safety functions are a basis for developing technical requirements imposed on design options. By identifying the functions that are to be performed in order to guarantee the post-closure safety of the repository, one makes a natural link between safety objectives, the features and processes that are critical as regards safety functions, and the engineering options that may fulfil safety functions. The near circular cross-section profile of the engineered structures, their dimensioning, their dead-end arrangement, their closure with low-permeability seals, the backfilling of all drifts and the choice of materials (concrete, steel, clay, bentonite) all indeed contribute to the three main safety functions.

With this approach, technical design solutions are presented for waste disposal packages, disposal cells, and for underground infrastructure. To assess the industrial realism of suggested design solutions, Andra has based its studies on existing industrial feedback, has conducted the design of underground facilities and operational equipment up to a reasonable level of detail, and has conducted specific tests (above ground), pertaining for example to the horizontal emplacement of C-type waste. Furthermore, studies related to operational safety were conducted, to cover a range of situations, such as fire in underground drifts. The next step to go will be to design, built and monitor in situ, full scale components and sub-systems, mimicking the future installations.

The pars (phenomenological analysis of repository situations): a key approach to structure the description of the repository evolution

As a support to the definition of architectures, and in close relationship with them, Andra has systematically described the possible pre- and post-closure situations, as compiled and presented in the pre- and post-closure phenomenological analysis of repository situations (PARS) [6,9]. It is based on the level of understanding of the undisturbed geological medium properties, for which it then systematically analyses all THMCR[1] transient impacts related to the presence of a repository, from construction until all transients have settled into a new thermal, hydrogeological and mechanical equilibrium (1 000 - 10 000 years). The design of the repository itself, (see Figure 3) divided into zones, aimed at limiting interactions between waste type (both chemical and thermal).

Figure 3. **General installations architecture**



To ensure thoroughness in describing the repository evolution and identifying the main occurring phenomena, the analysis uses a space and time breakdown of the repository components and their evolution during pre- and post-closure. The temporal evolution of the pre-closure PARS is structured according to the situations typical of the evolving, reversible disposal procedure. Emphasis was placed on the impact of progressive construction, operation, and closure of waste disposal cells, modules, zones, connecting drifts and shafts. In particular, commensurate with the technical breakdown of reversible management, the evolution of each component during the successive closure steps was analysed. For example, specific studies address the evolution of a C-type waste disposal cell before cell closure, after closure, after all cells in a given module are filled, closed, and the module itself was closed, etc. Comparable studies were conducted for B-type waste disposal cells, connecting drifts, etc.

1. THMCR: thermal, hydraulic, mechanical, chemical and radiological.

The temporal evolution of the post-closure PARS is structured according to the transient evolution of the repository – the thermal phase, the resaturation phase, a phase of gradual liner degradation combined with increasing mechanical load – and by a final "situation" describing the very long-term evolution. The actual durations depend in part on the type of disposal package considered. While the transient thermal period is bound by approximately a thousand years for C-type waste, it is expected to last several times longer for spent fuel. The resaturation phase may be prolonged by the production of hydrogen from anoxic corrosion of steel components.

Managing uncertainties: the QSA

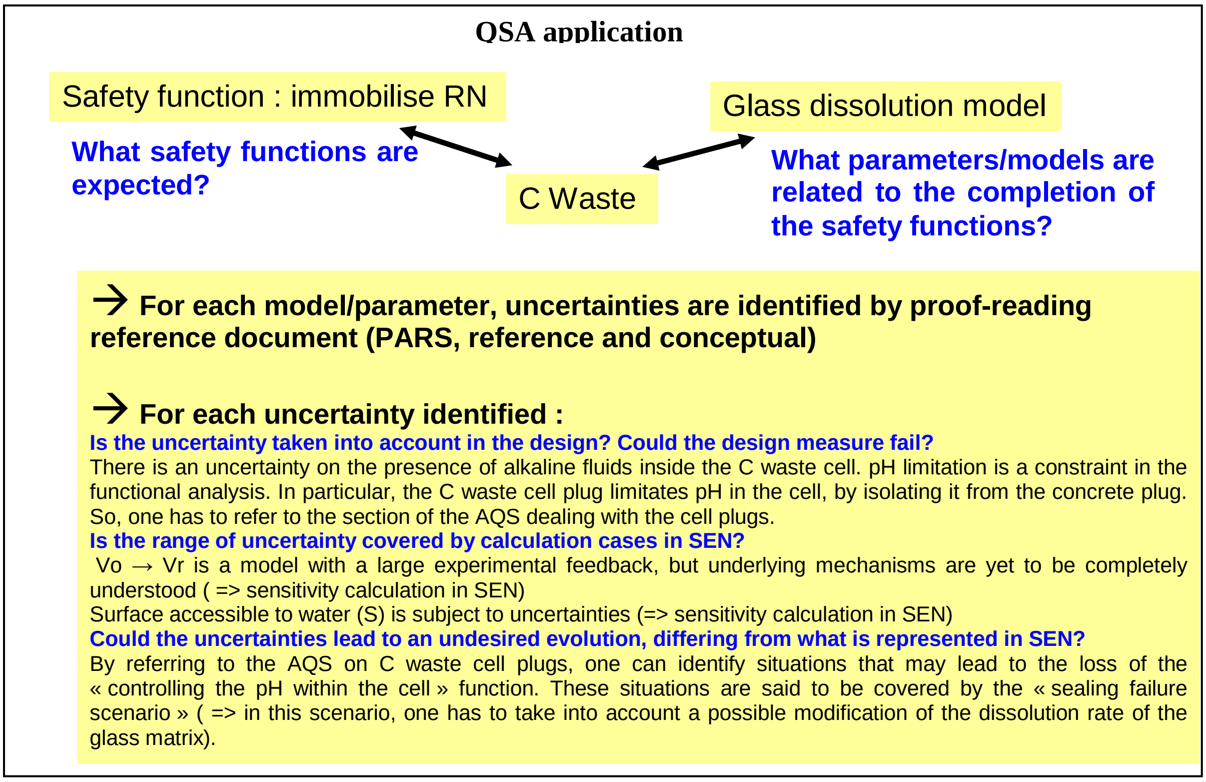
The qualitative safety assessment (QSA) consists in identifying uncertainties and studying their influence on repository evolution, thus analysing the limits of validity of the given scenarios. It systematically confronted the design options of each major repository component with the functional analysis, PARS and supporting simulation results. It makes it possible to highlight uncertainties significant with regard to safety. It then verified whether design options are robust in light of these uncertainties or not, the latter situation meaning uncertainties may affect the safety functions.

The QSA methodology was developed specifically for Dossier 2005 Argile. It was based on previous attempts and on the comments that these attempts generated, especially from the 2003 NEA peer review of "Dossier 2001 Argile". The aim was to provide traceability in the management of uncertainties. The reader of Dossier 2005 Argile, and especially safety evaluators, have a direct access to a list of the uncertainties that have been managed in the dossier, explaining how they have been managed and what consequences they might have on safety. This proved useful when discussing the management of uncertainties with the various evaluators. Those uncertainties are:

- Uncertainties on the initial data of the project as such waste inventory.
- Uncertainties on the characteristics of components as such measurement uncertainties, validity of the use of data taken from the literature, limitations due to changes of scales, limitations on the definition of the features (e.g. the notion of Kd)…
- Uncertainties on processes as such validity of models used to represent them, validity of the use of these models over very long timeframes…
- "Technological" uncertainties covering the reliability of components, their implementation and their quality control before implementations such as a poorly manufactured container.
- External risks as such seismic event;

With regards to the Functional Analysis and to the PARS, the QSA consisted in a much more systematic identification throughout scientific documentation (PARS, reference and conceptual notes) of uncertainties, by safety engineers who have not participated in the scientific work. The QSA analyses each uncertainty (on component's characteristics, its evolution, and its interaction with other components) that may either (i) affect its ability to perform a safety function, (ii) or have an influence on another component's ability to perform a safety function, or (iii) modify the component's environment in a way that could affect the way the component fulfils its functions. This analysis permits to check if the uncertainty is taken into account either by design or by the way the normal evolution scenario "SEN" it represented. In the framework of the dossier 2005, it allowed to identify the uncertainties that were accounted for by the SEN and the related sensitivity studies. The uncertainties that are covered by design or sensitivity studies are not easy to define *ex ante*. QSA allowed for an *ex post* analysis, and the identification of residual uncertainties that thus needed to be addressed especially by the means of the alterated evolution scenarios "SEAs". It therefore helped to check that the SEAs provided for an as complete as possible description of foreseeable altered evolutions. It also helped to define additional quantitative assessments by identifying sensitivity cases, and by shedding light on possible couplings of different uncertainties (see Figure 4).

Figure 4. **Illustrative example of QSA application**



Finally, the QSA offers an integrated vision of all uncertainties by taking into account the various types of treatment (qualitative, calculation results, and scenarios). In that context, a set of four "Altered evolution scenarios" (SEA) were developed to provide an understanding of the potential impact of unlikely future evolutions related to specific system failures: (i) partial or overall deterioration of seal performance, (ii) waste disposal packages failure (WPD), (iii) human intrusion and (iv) strongly degraded safety functions. As an end calculation, results (RN flows through barriers & end-of pipe impact) based on these SEAs and sensitivity cases within the SEN and SEAs make it possible to evaluate overall repository feasibility and robustness, with information on the contribution of each component/barrier to safety.

The scenarios and their evaluation

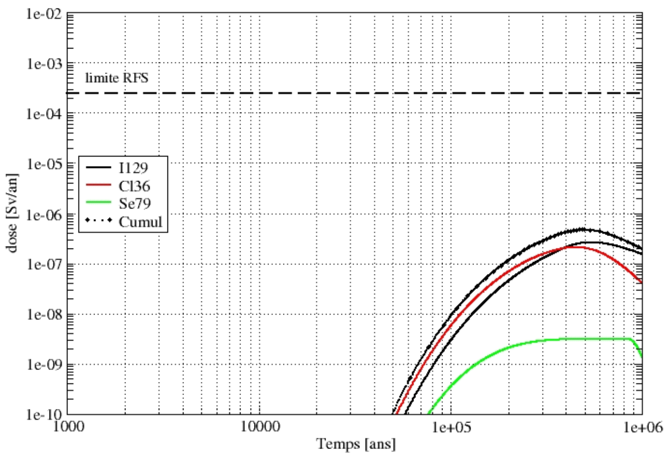
The basic Safety Rules, RFS III.2.f, require safety to be quantitatively evaluated by the means of "situations" and so as to avoid confusion with PARS, Andra uses the word "scenario" that encompasses all possible evolutions of the repository and that are judged as the most unfavourable in terms of consequences, among all possible evolutions that can be reasonably foreseen. "Scenarios" are simplified descriptions of the repository. The system representation for the safety model thus developed is based on a "Normal evolution scenario" (SEN), which purpose is to provide a bounding value for all likely or probable future evolutions. For example, the event of a few early waste package failures is included in its description. Calculation results based on this SEN are at the core of the performance assessment of the repository. Under the logic on which the Dossier 2005 is based, the altered evolution scenarios (SEAs) were first defined based on feedback from Andra's experience, analysis of situations taken into account internationally, and the recommendations of basic safety rule RFS III.2.f. The main types of situation to be covered and the main calculation cases were established on the basis of this definition.

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

Four altered scenarios were considered: the “waste package failure” scenario, the “seal failure” scenario, the “borehole” scenario, and the “severely degraded evolution” scenario (worst-case scenario).

The simulation results provide several indicators for evaluating safety. One such indicator is the comparison of the calculated effective dose against the dose objective of 0.25 mSv per year, recommended by the RFS.III.2.f for a normal evolution. For example, Figure 5 shows that the effective dose received by a critical group from all vitrified waste remains close to three orders of magnitude below this value. Peak dose occurs after about 500 000 years. The only radionuclides shown to eventually leave the host rock are I129, Cl36 and Se79 (Figure 5).

Figure 5. **Effective dose from HLW**



Other indicators allow assessing the performance of individual component with respect to their safety functions (for example, molar fluxes of radionuclides, which are independent of uncertainties on the future evolution of the biosphere). Among the analysed indicators are (i) the relation between convective and diffusive flux in the repository and the host rock, (ii) the overall activity leaving the waste packages, the underground structures and the host rock, as compared to the initial quantity contained in the waste packages, (iii) the activity flux at each of these components, (iv) and the concentration distributions of dissolved materials in the host rock and in surrounding formations.

Conclusion and way forward

The analysis shows that in all envisioned situations (normal or altered), the repository and surrounding host rock fulfil the three major safety functions, without relying excessively on any single component. In any event, the Callovo-Oxfordian host rock plays a major role in immobilising radionuclides and in delaying and reducing their migration to the environment. In all scenarios – even accidental or altered – the repository performance provides significant margins to the dose objective recommended by the RFS III.2.f. In conclusion, Dossier 2005 Argile supports, with both qualitative and quantitative arguments, the feasibility of a reversible and safe repository in Meuse/Haute-Marne Callovo-Oxfordian clay.

Way Forward: The 28/06/06 law [10] paves the way to go for siting and construction of a repository in the transposition zone of the Meuse/Haute-Marne laboratory. Future R&D must lead Andra to apply for a construction license in 2015, in order to be able to operate the repository ten years later, if the siting and licensing process is successful.

This law confirms geological repository as the reference solution, in combination with interim storage. It sets milestones and requirements:

- Retrievability is to be demonstrated by Andra, with conditions to be determined by a law in 2016. The conditions posed by this future law are sine qua non conditions to be issued a permit to operate.
- Andra still have to demonstrate safety and operability at a detailed level, using studies, experiments and full scale demonstrators, when possible.
- Uncertainties must be reduced, to meet the robustness and ALARA criteria.
- If issued a permit in 2016, Andra must be able to build the repository and operate it in 2025.

Acknowledgement

The authors acknowledge S.Mayer and F. Plas for their relevant comments on the paper.

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DEVELOPING A SAFETY CASE FOR ONTARIO POWER GENERATION'S L&ILW DEEP GEOLOGIC REPOSITORY

T. Kempe, P. Gierszewski, R. Heystee, M. Jensen, H. Leung
Ontario Power Generation, Canada

Introduction

Ontario Power Generation (OPG) is proposing to build a Deep Geologic Repository (DGR) for low and intermediate level radioactive waste (L&ILW) at the Bruce site in the Municipality of Kincardine, Ontario, Canada. The Canadian Nuclear Safety Commission (CNSC) has issued a draft scoping document for the Environmental Assessment (EA) required prior to licensing [1], and an associated CNSC public hearing took place on October 23, 2006 in Kincardine. Licensing and construction work is expected to take a further twelve years, leading to an in-service date of approximately 2017.

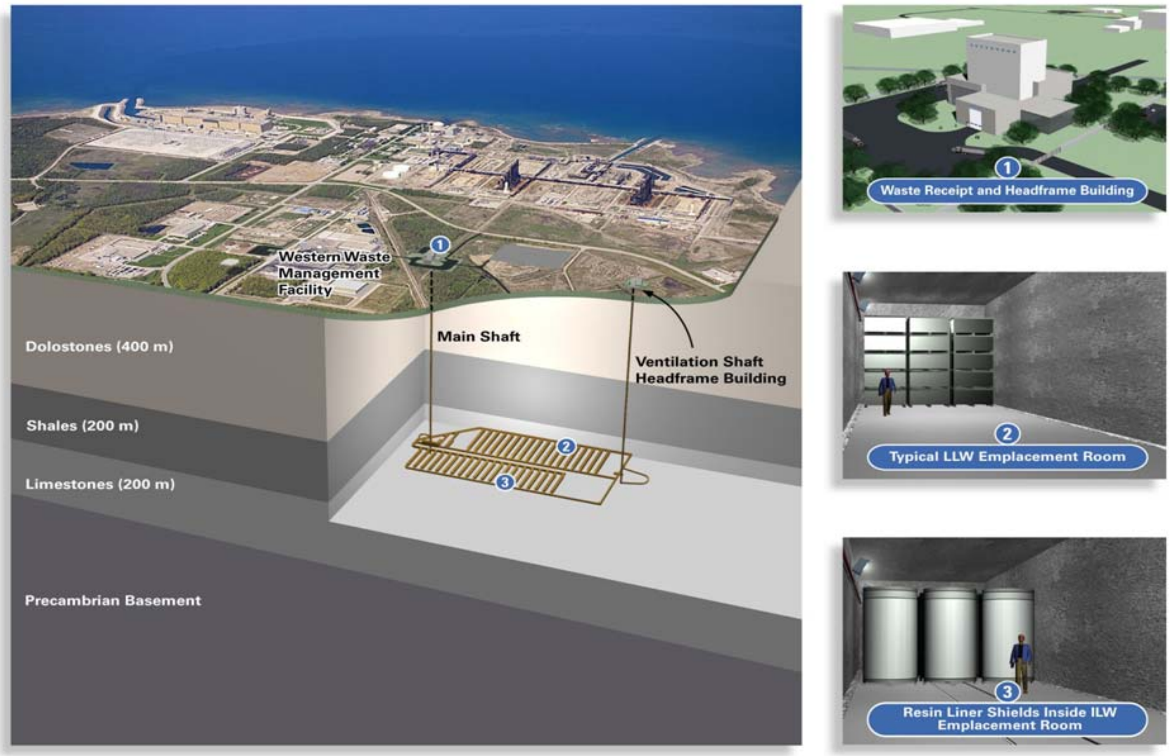
This paper presents a summary of the Safety Case for the DGR as currently developed. The purpose of this iteration of the Safety Case is to provide a common understanding of what needs to be demonstrated within the Safety Case for the EA stage of regulatory approval. This iteration is based on preliminary conceptual design information, general knowledge of the geologic setting, and scoping safety assessment calculations. These need to be verified – and revised if necessary – following site-specific investigations and more detailed and comprehensive design and analyses.

Overview of proposed Deep Geologic Repository

Figure 1 shows an artist's rendering of the current concept for the DGR in the sedimentary rocks of Palaeozoic age underlying the Bruce site. The repository would consist of a series of horizontal emplacement rooms, excavated in low-permeability argillaceous limestone, 660 m below surface. Access to the repository would be either by concrete-lined vertical shaft or inclined ramp (access by vertical shaft is shown in Figure 1 and is assumed in the remainder of this paper).

Following the operational period, shafts and boreholes would be sealed, and the surface environment at the site would be restored. Over thousands of years, the repository would gradually fill with porewater from the surrounding rock, and, potentially, from higher layers via the shaft seals. Small amounts of radioactivity would become dissolved in this water. In addition, gases would be slowly generated in the repository from waste degradation and corrosion. The post-closure safety case is based on the intrinsic quality of the geosphere at the site – its favourable flow system properties, its long-term stability, and its predictability. Together with careful design of shaft sealing systems, these properties ensure that radionuclide transport from the repository would be very slow, and as a result virtually all the radioactivity from the waste would decay within or near the repository.

Figure 1. **DGR concept at the Bruce site**



Regulatory context

Under Canada's National Framework for Radioactive Waste Management, waste producers are responsible for funding, organisation, management and operation of disposal and other waste management facilities [2]. For used fuel, the programme for long term management is set out further in federal legislation and is a national programme. For L&ILW, each waste producer decides on their programme for long term management, within the licensing system, guided by principles set out in CNSC's regulatory documents. The Nuclear Safety and Control Act (NSCA) and regulations provide that licences are required to prepare a site, construct, operate, decommission and abandon a nuclear facility such as the DGR. Before issue of a licence by CNSC, a decision statement is required on the acceptability of the proposed project under the Canadian Environmental Assessment Act. A single EA decision covers site preparation, construction and operation.

Further guidance is provided by CNSC's regulatory policies, standards and guides. Policy document P-290 [3] gives the high level expectations relevant to the safety case, in particular that:

- the assessment of future impacts of radioactive waste on the health and safety of persons and the environment encompasses the period of time when the maximum impact is predicted to occur, and
- the predicted impacts on the health and safety of persons and the environment from the management of radioactive waste are no greater than the impacts that are permissible in Canada at the time of the regulatory decision.

Detailed guidance setting out CNSC's expectations for the Safety Case is given in Regulatory Guide G-320 [4], and, for the EA stage, in the scoping document [1]. The DGR programme will in addition take into account applicable international guidance.

Approach to the Safety Case

Consistent with the NSCA and with CNSC's regulations and regulatory policies, the overall objective of long term radioactive waste management is to protect human health and the environment now and in the future. The specific safety objectives of the proposed DGR are as follows:

1. Isolation of the waste away from the biosphere.
2. Long-term containment of the waste to allow radioactive decay.
3. Retardation and attenuation of radionuclide migration to the surface.
4. Robust design and location to minimise uncertainty in long-term safety.

The DGR safety strategy has been developed consistent with the international Nuclear Energy Agency's Safety Case approach [5]. Key elements include stepwise planning and implementation, integration in the overall management strategy of technical work in support of the Safety Case, emphasis on the geosphere barrier, an iterative approach for development of technical studies, multiple safety functions contributing to meeting the safety objectives, structured analysis of the evolution of the system and of potential release mechanisms and pathways, simple robust arguments supported by multiple lines of reasoning including more detailed calculations, and consistency with international practice.

Current understanding of the site geologic setting, together with the results of preliminary safety assessment and conceptual engineering work, has allowed formulation of the following set of high level arguments contributing to the safety case:

- The site geoscientific conditions and features provide several independent lines of evidence regarding the setting, which together suggest that the safety objectives can be achieved with a high degree of assurance.
- The wastes are those safely handled at existing storage facilities. The repository can be built and operated safely using proven technologies.
- Post-closure dose estimates are very small because:
  - mass transport of contaminants through the host rock is diffusion limited;
  - construction of the repository will not change the overall diffusion-dominated environment;
  - earthquakes, glaciation or other natural events will not disrupt the repository;
  - gases generated by corroding wastes are safely retained, and disperse slowly; and
  - the repository is safe from inadvertent human intrusion.

These arguments will be tested and supported in ongoing work. Site characterisation, design, and development of the safety case are planned as an integrated, iterative process, with common numerical models and representations of the site linking geoscience and safety assessment. The expectation is that results from the first phase of site characterisation, conceptual design and safety assessment will be used to support the Safety Case presented for the EA review.

Siting of the DGR

The proposed location for the DGR is at the Bruce site in the Municipality of Kincardine, on the eastern shore of Lake Huron. The Bruce site has been the location of nuclear activities since 1960.

Currently, L&ILW from the Bruce, Pickering and Darlington nuclear generating stations in Ontario is processed and stored there, at OPG's Western Waste Management Facility (WWMF).

The choice of the Bruce site for the DGR resulted from an approach to OPG in 2002 by the Municipality of Kincardine, seeking to study options for long-term management of L&ILW. A consultant was contracted to conduct an Independent Assessment Study (IAS), considering geotechnical feasibility, safety, and potential environmental, social and economic effects of three options: enhanced processing and long-term storage, covered above-ground concrete vaults and a deep geologic repository [6,7]. While the IAS concluded that each of the options was feasible for some or all of the low and intermediate level waste, the Municipality indicated a preference for the DGR option as providing the highest level of safety, and the DGR was selected in April 2004 by resolution of Kincardine Council. In August 2004 the OPG Board approved the DGR proposal.

A Host Community Agreement was signed between OPG and the Municipality of Kincardine in October 2004 [8,9]. As one of the provisions of the Hosting Agreement, community consultation was conducted by Kincardine to gauge community acceptance of the proposed facility. A telephone poll of permanent and seasonal residents endorsed the proposal.

Communications with the host community, other local communities, and First Nations will continue throughout the project.

Wastes to be emplaced

The DGR will receive all L&ILW produced by the OPG-owned nuclear generating stations through the remainder of their operating life, as well as L&ILW currently in interim storage at the WWMF. Projected as-stored estimates of waste volume are approximately 130 000 m^3 of LLW and 30 000 m^3 ILW. These volumes include waste from planned refurbishment programmes.

LLW consists of industrial items that have become slightly contaminated with radioactivity and are of no further use, such as rags, protective clothing and hardware items such as tools. ILW consists primarily of used reactor components, and the ion-exchange resins and filters used to purify reactor water systems. From an operational point of view, the major nuclides are ^{60}Co (half-life 5.3 a), ^{3}H (12 a) and ^{137}Cs (30 a). Refurbishment waste includes removed reactor core components, which are associated with greater amounts of induced radioactivity. These radionuclides are firmly fixed within the material matrix. The most significant long-lived retube radionuclide is ^{94}Nb (20 300 a).

The total activity at the end of the DGR operational period is estimated as 16 000 TBq. ^{129}I (1.6×10^7 a) and ^{14}C (5730 a) are the radionuclides of most relevance in the long term safety assessment as they are long-lived and relatively mobile.

Engineering design strategy

As part of feasibility studies, a conceptual design has been developed for the DGR [10,11], illustrated in Figure 1. The design is consistent with experience in underground structures, such as limestone mines, in similar sedimentary formations in North America. This design will be further developed in ongoing work contracted with companies with relevant engineering and mining experience.

Geological repository design is an iterative process and the design may be modified based on:

- new data about the site generated during subsurface investigations, for example information related to rock strength, *in situ* stress magnitudes and orientation and bedrock bedding;
- the results of safety assessment, in particular the pre-closure safety assessment and occupational radiation dose and conventional safety considerations;
- design optimisation;
- further definition of the inventory and categories of waste to be emplaced, and
- the establishment of waste acceptance criteria for the DGR.

An important aspect of the DGR design strategy is that containment is provided by the rock mass and repository shaft seals, and there is no additional engineered containment. This is based in part on the expected low permeability and sufficient mechanical strength of the rock. The wastes are emplaced in a range of steel storage containers, as used for interim storage, with steel or concrete overpacks where required. Although the impact of backfill and/or conditioning of the waste will be explored in ongoing safety assessment, it is not currently planned to seek to optimise the design by use of engineered barriers, as current results predict impacts many orders of magnitude below regulatory criteria. This will be reviewed if later results indicate otherwise. Not using additional engineered containment is advantageous both for maximising monitoring and retrievability in the near-term, and for managing gas generation in the long-term. Design optimisation in accordance with CNSC guidance [4] will focus on several areas, including shaft design and sealing, facility location and layout, configuration of selected waste packages, underground waste package handling, and waste rock management.

Site characterisation

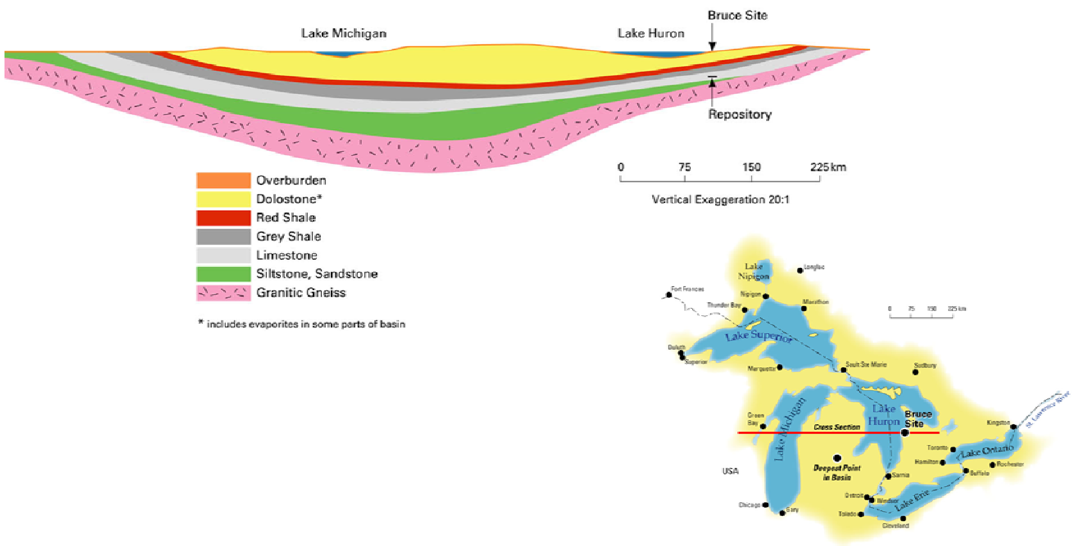
The bedrock underlying the Bruce site occurs within a well-known geologic feature referred to as the Michigan Basin (Figure 2). The studies carried out as part of the IAS [12], together with information compiled by Mazurek [13], indicate that favourable geological and hydrogeological conditions exist at the Bruce site, relevant to demonstrating the safety of the DGR, as follows:

1. The deep horizontally-layered shale and argillaceous limestone sedimentary sequence that will overlie and host the DGR is geologically stable, geometrically simple and predictable, relatively undeformed and of large lateral extent.
2. Active faulting and seismicity at and near the site are very limited.
3. The deep argillaceous formations that will host the DGR will provide stable and dry openings.
4. The regional stress regime (horizontally compressive) is favourable with respect to sealing of any vertical fractures and faults.
5. The deep shale and argillaceous limestones are thick and of very low permeability, providing a 200 m thick bedrock horizon for the waste management facility, and an additional 200 m thick barrier to upward migration from the facility.
6. The deep groundwater system in the shales and limestones is saline (about 100-200 $g \cdot L^{-1}$), stagnant, stable and ancient, not showing evidence of either glacial perturbations or cross formational flow or mixing.
7. The shallow water supply aquifer in the upper carbonate bedrock is hydrogeologically isolated and protected from the sluggish deep saline groundwater system.

The Geoscientific Site Characterisation Plan (GSCP) for the DGR is aimed at providing evidence to test the validity, or otherwise, of these assumed favourable characteristics, and also at providing the detailed data and understanding needed to support the Safety Case, to allow quantitative safety assessment for demonstration of compliance with acceptance criteria, and for design of the repository [14]. Site characterisation is complemented by studies aimed at developing a geosynthesis, or integrated geoscientific understanding of the past, present and future evolution of the Bruce site, and by studies and projects undertaken to build confidence in site suitability and the Safety Case. Independent oversight of the GSCP through development and implementation is provided by OPG's Geoscience Review Group (GRG), a group of internationally renowned scientists and engineers who, among other roles, ensure that information and lessons from similar geological repository programmes are reflected in site characterisation activities.

Implementation of the GSCP is now under way. A 2-D seismic survey was carried out in October 2006, and drilling of the first two deep boreholes started at the end of 2006. Other activities in this phase include installation of an enhanced borehole seismograph network to detect M-1 events within 40 km of the Bruce site, and refurbishment of existing on-site bedrock monitoring wells to establish baseline hydrogeologic conditions in the shallow aquifers to depths of 100 m.

Figure 2. **Geological setting of the proposed DGR within the Michigan Basin**



Consistent with the EA scoping document, OPG will consult with CNSC staff with regards to the adequacy of the subsurface characterisation data to support EA preparation in 2009.

Safety Assessment

The DGR safety assessment provides a quantitative measure of performance to demonstrate compliance with radiological protection and other criteria. The safety assessment work is aimed at carrying forward the understanding provided by geoscience into an examination of the overall system, including potential disturbance caused by the repository, and of the pathways by which radionuclides and non-radiological contaminants may reach the accessible environment. An approach following the IAEA's ISAM safety assessment methodology [15] has been adopted. This methodology encourages a well-structured, transparent and traceable approach. In addition, within the overall iterative structure of the technical studies, safety assessment follows an iterative process, with the results from each iteration used to guide further development work.

The main postclosure safety assessment scenarios of interest are the Reference Scenario, the Human Intrusion Scenario, and Disruptive or failure scenarios

The *Reference Scenario* considers the likely evolution of the site, the repository and the waste. Analysis cases include a constant climate and biosphere, and a climate and biosphere which evolve due to glaciation. Radionuclide movement through the limestone and shale layers would take hundreds of thousands, or millions, of years, and most of the radionuclides from the L&ILW would decay to insignificant levels before they moved even metres from the repository. The only radionuclides of potential concern are ^{129}I and, to a smaller extent ^{36}Cl and ^{99}Tc, because they are potentially mobile and long-lived. Doses calculated in the scoping safety assessment for LLW are many orders of magnitude below criteria [16,17].

The slow degradation of the wastes and the waste packages would also result, over hundreds or thousands of years, in the formation of gases, mostly H_2, CO_2 and CH_4, which contain radioactivity, mainly ^{14}C and ^{3}H. The repository is not backfilled, so there is a large void volume into which these gases could expand, and they are predicted to be retained safely within the DGR due to the favourable properties of the host rock. Even if the gas were to be released from the repository as it is produced, estimated dose consequences are low, because of the slow gas generation rate and dispersion in the upper 400-m groundwater system and atmosphere.

The *Human Intrusion Scenario* considers the hypothetical inadvertent disruption of the wastes in the future, assuming memory of the site had been lost, and essentially bypassing the geosphere barriers. While the likelihood of any intrusion would be very small, in order to demonstrate the robustness of the DGR, a stylised human intrusion scenario is considered. The scenario examined by Quintessa [16,17] for LLW considered extraction of borehole samples that contain waste. The limited amount of waste that would be retrieved in this scenario means that the calculated dose rates are very low.

Disruptive/failure scenarios to be considered in ongoing safety assessment may include seismic events, undetected fracture outside the immediate site area, unsealed (open) borehole, degraded shaft seals, and variability in the permeability and sorption characteristics of the surrounding rock. These scenarios include “what-if” cases aimed at exploring the robustness of the system.

Pre-closure safety assessment is also in progress. Pre-closure assessment considers the potential impact on the public, environment and workers during repository operation, decommissioning and closure. The safe operation of the WWMF, of mines, and of other geologic repositories, provides confidence that the preclosure operation of the DGR would be safe.

Summary and conclusions

Development of the Safety Case for the DGR is founded on an assumed site descriptive model having a number of favourable features contributing to long-term isolation, containment and retardation. Over several years, an integrated, stepwise programme of geoscientific site characterisation and complementary studies, linked to safety assessment, will be used to test and refine this model and to build confidence in the Safety Case.

Understanding of the DGR setting and evolution developed to date, together with the results of preliminary assessments, gives confidence that the site possesses favourable geological and hydrogeological characteristics for isolation of the waste from the biosphere and near-surface environment, that there are a number of complementary arguments supporting the conclusion that isolation will be achieved, and that robust safety assessment can be carried out demonstrating that the proposed DGR will meet regulatory criteria for protection of human health and the environment.

No outstanding issues with the potential to compromise safety have been identified, and the DGR programme is now moving forward with detailed site characterisation and with development of the studies and analyses needed for the EA review process.

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DEVELOPMENT OF A METHODOLOGY FOR AN ENVIRONMENTAL SAFETY CASE IN THE UNITED KINGDOM

L.E.F. Bailey
United Kingdom Nirex Limited

Introduction

Following an extensive consultation by the independent Committee on Radioactive Waste Management (CoRWM), the UK Government has endorsed deep geological disposal as the preferred long-term management solution for higher-level radioactive wastes in the UK. As yet there is no chosen site for a geological repository. The Government has indicated that it wishes to pursue a volunteer approach in which local communities will be invited to consider whether they might wish to host such a facility.

The UK has intermediate level wastes (ILW) and high level wastes (HLW) for disposal and must consider the possibility that other radioactive materials, such as spent fuel (SF), may subsequently be declared as wastes. It has not yet been decided whether these wastes will be disposed of in the same or neighbouring facilities (co-location), or whether different facilities will be constructed at different sites for ILW and HLW/SF.

Before any such facility can be implemented it will be necessary to demonstrate that there will be no significant adverse health or environmental impacts and that all the necessary regulatory criteria can be met. It is therefore proposed to develop an Environmental Safety Case (ESC) that will draw together the arguments and analyses that build confidence in the safety of deep geological disposal. The ESC will address the operational, transport and post-closure safety and environmental implications of a geological repository. It will consider disposal of both ILW and HLW/SF.

Initially the ESC will be based on generic repository concepts and locations. The ESC will evolve with the repository development programme in a staged way. It may be used to inform a site selection process, support permissions for detailed site investigation work, and eventually form part of a regulatory submission to proceed with development of a repository.

Aims and Audiences for the ESC

The ESC will be broader than a traditional performance assessment. It will draw on evidence from a wide range of sources to demonstrate why a geological repository would be safe over all timescales. The focus will be on building confidence in safety, rather than just calculating the risks associated with geological disposal.

The audiences for the ESC will include:

- The potential host communities.

- The regulatory bodies that provide the necessary licences and authorisations for the various stages of implementing a repository, as it is envisaged that the ESC would form the basis for formal regulatory submissions.
- Those who advise Government ministers on repository implementation plans.

These audiences are likely to have different needs, particularly in terms of the level of technical detail that they will wish to see in an ESC. It is a particular challenge for the ESC to be able to address these different needs. To achieve this, it is proposed to present the ESC as a linked hierarchy of documents. The top-level summary report would not require the reader to have detailed scientific knowledge. This report would reference underpinning reports that would take the reader to increasing levels of detail, including assessment reports and research studies. The aim is that all information used in the ESC would be traceable back to its source, enabling the reader to follow through any aspect of the safety case or strand of reasoning.

It is one of the regulatory requirements [1] that the safety case should not depend unduly on any single component. The multi-barrier disposal concept ensures that safety does not rely solely on any physical part of the repository design, but rather on a combination of barriers and associated safety functions working together. It is also necessary to demonstrate that the safety case does not depend on any single safety argument. Therefore the approach to demonstrating safety involves multiple lines of reasoning, drawn from a variety of sources and information types.

These multiple lines of reasoning include both qualitative and quantitative arguments:

- Qualitative arguments may be drawn directly from research or from comparisons with familiar situations or well-established processes. Such arguments are most valuable in explaining the processes that affect how the repository system evolves.
- Quantitative arguments, often produced by computer modelling, are important for demonstrating compliance with the numerical aspects of the regulatory guidance. In particular, calculations are required to demonstrate compliance with the radiological risk target [1].

Whereas there is considerable international experience of building a safety case around quantitative arguments, there is perhaps less experience of developing a broader safety case. It is a particular aim to integrate a wide range of data into the ESC, so that the detailed mathematical analyses are supported by more tangible safety arguments. For example, comparisons with natural and anthropogenic analogues can help to build confidence in processes that have occurred over very long timescales.

Proposed ESC Methodology

The proposed approach for the ESC is to start by describing the variety of geological repository concepts that have been developed around the world and explain the extent to which these would be suitable for the radioactive wastes existing in the UK and could be adapted to the various geological environments found in the UK.

The evidence to support the safety of these concepts in a UK setting will be largely qualitative at this stage. Hence the ESC will be based on a foundation of generic safety arguments to support confidence in the long-term safety of deep geological disposal. These arguments will relate to the time over which it is necessary to isolate the wastes from the accessible environment; the processes that operate over these timescales, both favourable and unfavourable; the extent to which we are confident in our knowledge of these processes and the implications for the safety of a repository system.

It is proposed that the ESC will also present “worked examples” for selected concepts. For each worked example the concept will be examined in more detail. This will include using a concept design to undertake numerical modelling work in order to calculate the risks involved in constructing and operating such a facility and the long-term safety and environmental implications. A variety of safety indicators will be discussed, both qualitative and quantitative.

At a later date, the ESC would be updated to reflect the conditions at specific site(s). A key role for the ESC at this stage might be as part of an application to obtain planning permission to start geological investigations at one or more sites. The assessments presented in this ESC would therefore need to be at a sufficient level of detail to enable decisions to be made about the likely suitability of the site(s) being considered for a geological repository. The regulatory and planning authorities, as well as the local communities would be key audiences for such an ESC.

Subsequently, the numerical assessments within the ESC would be updated as research and site characterisation studies provided more data concerning the specific site and repository design. Whilst the numerical aspects of the ESC will need periodic updating as the repository development programme progresses, the underpinning safety arguments for our confidence in the long-term safety performance of the natural and engineered barriers of a deep geological repository should be robust over the evolution of the ESC. This means that the ESC can be developed in a flexible way, taking account of programme decisions and the availability of new research data, whilst still maintaining and building upon the underpinning safety arguments.

Use of Natural Analogues in a Safety Case

Most stakeholders, including many with a technical background, have no real familiarity with processes operating on very long timescales, i.e. the geological timescales associated with deep geological waste repositories. In order to provide confidence that a repository will remain safe millions of years into the future, it is helpful to set out some of the evidence and reasoning that have led to an understanding of events and processes in the distant past. This is where it is recognised to be helpful to refer to natural analogue evidence.

Natural analogue studies focus on materials and processes in natural and archaeological systems which have specific affinities with different aspects of likely designs for actual repositories and the isolation barrier systems they incorporate. Natural analogue evidence has been built up over many years of scientific study of the Earth’s history. The EC NAnet project [2] has identified and catalogued over 70 natural analogues of relevance to geological repository systems.

Analogues can be regarded as natural experiments started thousands or millions of years ago, from which we are now collecting the results. Of course, as we ourselves did not design the experiments, it is unlikely there will be a perfect match to the repository conditions we wish to study. This must be taken into account when using natural analogue data. However, by giving us the results of experiments conducted over timescales we could never achieve in a laboratory, natural analogue studies are one of the most useful tools we have to increase our understanding of the processes that control the evolution and safety of a repository over time.

It can be difficult to extract precise numerical data from complex natural systems due to uncertain boundary conditions. Therefore natural analogues are primarily used to provide qualitative information to help develop or confirm conceptual models. They can be used to identify the processes responsible for the evolution of a natural system, confirm the spatial and temporal scales over which the processes operate and the extent to which different processes are coupled. This means that natural analogues can be used to test whether significant features, events and processes (FEPs) have been included in performance assessment models.

Natural analogues should not be viewed in isolation. They are complementary to field and laboratory investigations. The main disadvantages of laboratory studies are that they can only be conducted over a relatively short timescale and they often relate to simplified, unrealistic systems. Natural analogue studies do not have these disadvantages, but their drawbacks relate to uncertainties over the boundary conditions and the fact that they may not relate exactly to the conditions in a repository. By presenting arguments drawn from both well-defined field and laboratory experiments, and natural analogue studies, it is possible to build greater confidence in our understanding of the evolution of a repository system.

Presenting the Safety Arguments (Timeframes-based Approach)

Our confidence in the stability and safety of different components of a repository system varies over time. With time, radioactive decay reduces the risk associated with the wastes. However, the uncertainty associated with the performance of the safety barriers may increase over time. Understanding the stability of the different safety barriers, and hence the timescales over which they can be relied upon to provide the necessary safety, is fundamental to the construction of a safety case.

It is for these reasons that it is proposed to use a timeframes-based approach to present the safety arguments in the ESC. The timeframes are defined in terms of the main safety barriers and functions of the repository system. Once a repository is closed the safety barriers are all in place. However, their relative importance will vary over time. For example, the geological barrier will be present throughout the lifetime of the repository, but it may take several thousand years before any radionuclides reach the geosphere. During the earlier period it is the engineered barriers that are most important in containing radionuclides. The safety barriers and functions can therefore be mapped to different timeframes, over which their safety function is at its most important. These timeframes are nested, rather than sequential, reflecting the fact that the safety functions operate in parallel, complementing each other, for much of the assessment period.

As the safety barriers and functions generally relate to different components of the repository system, there is also an aspect in which the safety barrier-defined timeframes can be regarded as representing different spatial, as well as temporal, scales of the repository system. The assessment of each timeframe therefore tends to be focussed at a specific spatial as well as temporal scale. This enables safety assessment models for each timeframe to be tailored at an appropriate level of detail (for example representing greater spatial detail, down to the performance of individual package types, in the early timeframes, which would not be appropriate when the focus is radionuclide transport through the geosphere, at later times).

Natural analogue studies are most readily mapped to specific timescales, as they refer to events or processes that have happened in the past on a known timescale. Specific timescales for analogue studies include:

- Historical – the timescale over which historic records are available (in the UK, the existence of the Domesday book, written in 1086, suggests the historical timescale may be up to around 1 000 years).
- Archaeological – the timescale over which anthropogenic evidence is available (maybe up to a few tens of thousands of years).

- Geological – very distant timescales for which evidence is only available from natural systems (can extend many millions of years into the past).

In terms of using analogue studies to support confidence in the performance of the safety barriers and functions of a repository system, analogues on the historical and archaeological timescales are most relevant for understanding the evolution of the engineered (man-made) barriers, whereas geological analogues can be used to build confidence in the long-term performance of the geosphere. In each case it is necessary to explain how evidence from the past can be used to support understanding about the future.

If a safety case is developed around appropriately defined timeframes, a simultaneous focus can be given to:

- the main safety functions and barriers;
- the timescales over which they are most important;
- natural analogue evidence relevant to the safety functions and their timescales of operation; and
- development of models on appropriate spatial scales to represent the safety functions.

This paper now considers how this approach could be applied to a generic example concept of ILW encapsulated in cement and packaged in stainless steel containers, surrounded by cement-based backfill, in vaults at several hundred metres depth in a stable geological environment. The timeframes for such a repository concept may be defined as follows:

- Timeframe 1: **Containment**. The waste container is mechanically and structurally intact. Only gaseous releases (via container vents) are possible, all other materials are completely contained within the waste packages. Institutional control of the repository site prevents inadvertent human intrusion.
- Timeframe 2: **The Package**. The physical containment afforded by the waste packages, including the wasteform itself, continues to retard the release of radionuclides by the groundwater pathway, even though localised corrosion may have reduced the integrity of some containers.
- Timeframe 3: **The Chemical Barrier**. The release of radionuclides continues to be retarded by the reducing, alkaline conditions established in the repository backfill porewater.
- Timeframe 4: **The Geological Barrier**. The geological barrier provides a long travel time to the surface, gives substantial dispersion and dilution and retards sorbing radionuclides. This prevents most radionuclides that leave the near field from returning to the surface environment and ensures that any radionuclides that do reach the surface do so in very low concentrations that do not pose any significant health risk.
- Timeframe 5: **Continuing Safety**. The long-term stability of the geosphere continues to provide safety at very long times in the future, even under significant external change. The physical presence of the geosphere still provides an isolation barrier at depth, even though changes in surface environmental processes mean that presenting radionuclide transport calculations is no longer appropriate.

Timeframe 1: Containment

The most important safety arguments in this timeframe relate to the performance of the waste container, and to demonstrating a good understanding of the behaviour of any gas released by the wastes.

The containers used for ILW in the UK are generally made from stainless steel. Evidence for the corrosion properties of iron-based alloys is available from a range of sources, for example:

- laboratory and field experiments;
- in-service information;
- archaeological artefacts;
- occurrences of native iron; and
- meteorites.

Analyses of these data show that, for iron corroding within a solid medium such as soil or concrete, the data can be rationalised in terms of a so-called "parabolic" equation in which metal loss is proportional to the square root of time [3]. This applies for up to a few thousand years at least; and if good meteorite data can be obtained it may be possible to extend this timescale.

The corrosion behaviour of stainless steel is dominated by the passivating effect of the chromium oxide layer that quickly forms on its surface. In-service data for stainless steel come primarily from buildings and structures erected in the first decades of the twentieth century (for example, the Chrysler Building in New York). Estimates of their corrosion rates are in good agreement with laboratory measurements. During underground storage/disposal of stainless steel radioactive waste containers, uniform corrosion rates are expected to be so low that, extrapolating from laboratory measurements, the stainless steel containers can be confidently expected to survive for many thousands of years.

In more aggressively corrosive environments, localised corrosion of stainless steel, e.g. pitting, can also occur. This is less predictable than uniform corrosion but its essential characteristics are well described by electrochemical corrosion theory.

During any interim storage of radioactive waste above or below ground, the waste packages would be kept in an air environment where localised corrosion can be controlled through the management of the storage facility environment – specifically, by limiting temperature, humidity and airborne salt and dust. After backfilling, the cement grout around the waste containers would provide a strongly alkaline aqueous environment in which the likelihood of localised corrosion is further reduced. Subsequently, oxygen would be consumed and the steel-grout surface would become anaerobic. Under these conditions it is likely that localised corrosion would no longer be possible unless oxidising species become available through other means, such as radiolysis of water or galvanic effects.

In summary, the repository conditions are designed to ensure significant, localised corrosion is unlikely to occur. Laboratory and analogue evidence provides confidence that with only uniform corrosion, the containers could have lifetimes of thousands of years. During this containment period it can be calculated that the levels of radioactivity in the wastes will decay to around 1% of their initial level.

Timeframe 2: The Package

Safety in Timeframe 2 largely relates to the stability of the waste package. In addition to the safety arguments for the resistance of the container to corrosion, given above, it is now appropriate to consider the physical stability of the waste encapsulation grout. This grout is typically based on Portland cement. It plays an important role in limiting the transport of radionuclides within the waste package.

Archaeological analogues [4] for the longevity of the physical properties of Portland cement include:

- ancient lime mortars, such as that used in Hadrian's Wall in the UK;
- concretes used for Roman baths, which retain their low permeability; and
- still-intact Roman concrete buildings, such as the Pantheon in Rome, which show long-lasting strength.

These archaeological analogues suggest that cement and concrete are stable materials, over at least a couple of thousand years, thus supporting their use as isolation barriers in many disposal concepts.

Timeframe 3: The Chemical Barrier

In Timeframe 3 it is assumed that radionuclides have been released from the waste packages and it is now the chemical barrier (i.e. the vault backfill) that plays a key role in containing many of the radionuclides. It is therefore important to provide evidence of the properties and stability of the chemical barrier.

The Maqarin site in Jordan [5] provides an analogue for the longevity of the chemical properties of the cement backfill and evidence to support the cement leaching/groundwater buffering model used in safety assessments. Maqarin is a site of natural cement minerals, produced as a result of high temperature/low pressure metamorphism of clayey limestones. Its formation is dated at between 600 000 and 150 000 years ago.

Studies at Maqarin of the interaction of groundwater with cement minerals and the subsequent leaching of hyperalkaline groundwater and its interaction with the surrounding rock, indicate that the chemical conditions in a cementitious repository will exceed pH 12 and remain hyperalkaline for thousands of years, buffered by the dissolution of portlandite.

Studies at Maqarin have also helped us to understand the nature of the alkaline disturbed zone (ADZ), i.e. the region of rock around the repository affected by the hyperalkaline groundwater plume. Maqarin provides evidence for the creation of new cement phases along the flow path, which have the beneficial effects of closing flowing fractures and increasing sorption. This behaviour in a repository system would help to reduce the migration of radionuclides away from the repository. Sampling for microbial species in the natural hyperalkaline waters at Maqarin revealed only low number of microbes. Likewise, there was no evidence for the formation of additional colloids. These results help to build confidence that the hyperalkaline groundwaters do not introduce FEPs (in this case the creation of microbes or colloids) that could have an adverse affect on safety.

Timeframe 4: The Geological Barrier

The focus of this timeframe is the groundwater pathway through the geosphere. Relevant safety arguments relate to understanding the various functions of the geosphere which limit the rate at which those few radionuclides that leave the engineered barriers migrate from the repository to the surface. A good geological barrier provides the following functions:

- a long groundwater travel time from the repository depth to the surface;
- reduction of radionuclide concentration in groundwater by sorption to mineral surfaces or incorporation within newly formed minerals (mineralisation);
- dilution of the radionuclide plume, through mixing with uncontaminated groundwaters.

It is not easy to determine exactly the rate at which groundwater moves in the geosphere, not least because this can vary dramatically between different types of rocks. However, if upward advection were generally rapid, then hot waters from depth would be found commonly at the ground surface. In the UK warm water springs are very rare and are only known from three areas, Buxton-Bakewell-Matlock, Bath-Bristol, and Taffs Wells [6]. This confirms that rapid upward groundwater movement is rare in the UK. Measurements of groundwater isotopic composition can be used to determine groundwater age, which in turn indicates the groundwater travel time. (Comparisons of Oxygen^{18} and Oxygen^{16} isotope levels can be used to determine groundwater age in a similar way to radio-carbon dating.) Any future site selection process for a repository would place considerable emphasis on demonstrating slow groundwater movements.

Sorption of various radionuclides may be measured in short-term laboratory experiments [7]. Numerous measurements of the long-term sorption behaviour of some radioelements under natural conditions have been acquired as part of the natural analogue studies of uranium ore bodies, for example at El Berrocal and Palmottu [8] demonstrating that sorption can retard some radionuclides on geological timescales. Mineralisation reactions are known to occur in nature and have been reported from studies of natural uranium ore bodies, for example at Oklo in Gabon [9], and from the ADZ analogue at Maqarin [5].

There is considerable precedence from contamination events in shallow rocks that groundwater flow, in both fractures and the rock matrix, leads to spreading and dilution as the flow exploits numerous alternative pathways through the rock. An example of such a plume is that which developed around the Bowden landfill site in Ontario [10]. Similar processes are expected to occur as contaminated groundwaters move away from the repository. This will result in a plume of contaminants that will slowly disperse and dilute with increasing time and distance from the repository.

In considering natural analogue evidence for groundwater processes, it is important to ensure that the geochemical conditions of the analogue site are appropriate to those at the repository. Factors which may reduce the effect of retardation include the presence of organic complexants, colloids and micro-organisms. As noted in the discussion of the Maqarin site, analogue studies may also be helpful in determining the likely levels of such species.

Timeframe 5: Continuing Safety

The safety arguments for this timeframe relate to the enduring performance of the geological barrier in isolating the waste. By studying the geological response to past major external changes, such as seismicity, it is possible to understand how the geosphere may respond to future changes.

This timeframe extends to a million years in the future and beyond, by which time the radioactivity in the wastes will have decayed to about 0.01% of its initial value. Therefore, it is not proposed to present detailed numeric assessments for this timeframe. Instead a range of complementary indicators, safety arguments and bounding analyses will be used to provide assurance of the continuing isolation function provided by the geological barrier.

Natural analogue studies that are relevant to this very distant timeframe include those at Cigar Lake in Canada [11]. Cigar Lake is one of the richest uranium ore deposits found in the world. It lies at a depth of 450 metres (similar to that envisaged for a repository) and has been subject to the same processes that could dissolve radionuclides in a repository and return them to the surface. However, studies show that the high levels of radioactivity at Cigar Lake have had no effects on the surface environment, even though Cigar Lake has existed for thousands of millions of years.

However, perhaps the most powerful safety arguments that relate to this timescale are those that demonstrate the very long-term stability of the geosphere. At a well-chosen location, the repository is likely to be constructed in rocks that have seen little or no change over many millions of years. With an appreciation of such long geological timescales, it becomes possible to have confidence in the further stability of the geosphere for another million years or so.

Summary

In summary, the main benefits of the proposals for the ESC are regarded as follows:

- A multiple-factor safety case that draws on a wide range of safety arguments and analyses to build confidence in safety.
- A safety case that communicates to different audiences, including regulators and the general public.
- A safety case that builds an appreciation of the evolution of the repository system over different timeframes and describes those timeframes in the context of historical, archaeological and geological evidence – in particular giving an understanding of the very long-term stability of geological systems.
- A flexible approach that enables the ESC to develop to address the different stages of a repository development programme.

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STRATEGY FOR SAFETY CASE DEVELOPMENT: IMPACT OF A VOLUNTEERING APPROACH TO SITING A JAPANESE HLW REPOSITORY

K. Kitayama, K. Ishiguro, M. Takeuchi, H. Tsuchi, T. Kato, Y. Sakabe, K. Wakasugi
Nuclear Waste Management Organisation of Japan (NUMO), Japan

Abstract

NUMO's strategy for safety case development is constrained by a staged siting approach, which has been initiated by a call for volunteer municipalities to host the HLW repository. For each site, the safety case is an important factor to be considered at the selection steps – which narrow down towards the preferred repository location. This is particularly challenging, however, as every site requires a tailored repository concept, with associated performance assessment and an individual site evaluation programme – all of which evolve with gradually increasing understanding of the host environment. In order to maintain flexibility without losing focus, NUMO has developed a formalised tailoring procedure, termed the NUMO Structured Approach (NSA). The NSA guides the interaction of the key site characterisation, repository design and performance assessment groups and is facilitated by tools to help the decision making associated with the tailoring process (e.g. a requirements management system) and with comparison of siting and design options (e.g. multi-attribute analysis). Pragmatically, the post-closure safety case will initially emphasise near-field processes and a robust engineering barrier system, considering the limited geological information at early stages. This will be complemented by a more realistic assessment of total system performance, as needed to compare options. In addition, efforts to rigorously assess operational phase safety and the practicality of assuring quality of the constructed engineered barriers are components of the total safety case which are receiving particular attention now, as they may better discriminate between sites while information is still limited.

Introduction

The Japanese siting process for a vitrified high-level radioactive waste (HLW) repository comprises three steps, as defined by "The Specified Radioactive Waste Final Disposal Act " (hereafter "the Act"), promulgated in June 2000. In the first stage, Preliminary Investigation Areas (PIAs) for potential candidate sites are nominated, based on area-specific literature surveys focusing on the long-term stability of the geological environment. Detailed Investigation Areas (DIAs) are then selected from the PIAs, following surface-based investigations carried out to evaluate the key characteristics of the geological environment. In the final stage, detailed site characterisation, including studies in underground experimental facilities, leads to the selection of a site for repository construction.

The complex geological setting and active tectonics of the Japanese archipelago bring challenges for the repository implementer; even though the fundamental feasibility of safe disposal has been demonstrated in the generic H12 study (JNC, 2000), siting such a repository is a sensitive issue. In particular, acceptance by host community will play a key role, which led NUMO to decide on a process of open solicitation of volunteer municipalities, which was announced in December 2002.

The staged siting procedure associated with the volunteering approach imposes constraints on the safety case development strategy. In addition, the time plan for the early siting stages is very tight compared to most other national programmes. Initiation of the process, involving only literature surveys, is dependent on qualified volunteers coming forward. The literature survey stage leading to the selection of PIAs would run in parallel for all qualified volunteers over a relatively short time period. The Preliminary Investigation (PI), at which point field characterisation is initiated, is also relatively short and forms the basis for selection of a smaller number of DIAs, where underground characterisation facilities are constructed. The safety case is one of key factors to be considered at each selection step; however, its development is a challenging process because every site requires a tailored repository concept with associated performance assessment and an individual site evaluation programme. There are thus a number of clear decision points where sensitive choices must be made between alternative sites and associated designs at an early stage – under time pressure and based on limited data. NUMO is committed to making such decisions in an open and transparent manner, which will be aided by a formal programme development process.

The time plan is more extended – comparable to those in other national programmes – at the later stages of site characterisation and subsequent construction and operation phases. However, some particular characteristics of the Japanese case need special consideration – in particular, the relatively large inventory and high emplacement rate for the reference case. Operational practicality and safety are thus key aspects of any design and need special consideration in the total safety case (Kitayama *et al.*, 2006).

In order to maintain flexibility without losing focus, allowing the project to be implemented in a systematic manner despite such constraints, NUMO has developed a formalised tailoring procedure (e.g. Kitayama *et al.*, 2005a,b), termed the NUMO Structured Approach (NSA). The NSA was originally established to help the repository design process, but it guides the key coordination of site characterisation, design and performance assessment (PA) during the staged processes and also facilitates safety case development. This paper presents NUMO's strategy for safety case development, with particular focus on early stages of the siting programme.

Key elements defining the NUMO safety strategy and assessment basis

Siting factors

After a qualified volunteer comes forward, the staged siting starts with the selection of PIAs. The goal of selecting PIAs is to identify areas for conducting preliminary investigations, excluding those that would clearly be unsuitable as repository sites, based on data and information obtained through literature surveys. Japan lies in a region of active tectonics, characterised by dynamic geological processes and events such as volcanism and earthquakes. One of the first things that needs to be ensured is that a repository is not located where it could be adversely affected by such features – which is a requirement specified in the Act. This assurance has to be valid over a long period of time into the future, which means that account needs to be taken of the driving tectonic mechanisms and how these change with time (NUMO, 2004a). To make this clear, NUMO has published the "Siting Factors for the Selection of Preliminary Investigation Areas" (NUMO, 2002) which discusses how the factors that determine site suitability are determined. These factors are effectively the criteria that will be used to determine the acceptability of a volunteer – or to compare volunteers if there are too many of them to go through to the PI stage. The siting factors provide guidance and constraints for safety case development, specifically focused on geological stability in the early stage of repository programme. Siting factors for the later siting stages will be developed and have a similar role for future safety case development.

The Siting Factors for selection of the PIAs (Table 1) consist of Evaluation Factors for Qualification (EFQ) and Favorable Factors (FF). The EFQ are factors that assess compliance with legal requirements and define specific assessment criteria. They relate to geological stability, including potential volcanicity, seismicity, rock deformation and faulting, and land uplift/erosion. The FFs, which cover geological, geographical, environmental, and social aspects, are used to assess characteristics of PIAs comprehensively (and comparatively if necessary) for areas where compliance with the legal requirements has been confirmed on the basis of the EFQ. The assessment of the FFs will be used as input to define the PIA-specific characterisation programme and repository design study. After these basic checks using the siting factors, it is important to determine more quantitatively whether it would be practical and safe to develop a repository in a given siting environment.

Table 1. **Siting factors for selection of the PIAs (from Kitayama, 2006)**

Siting Factors for Qualification (EFQ)	
Exclusion based on	
– Clearly identified active faults	
– Within a 15 km radius of the centre of Quaternary volcanoes	
– Uplift of more than 300 m during the last 100 000 years	
– Unconsolidated Quaternary deposits	
– Economically valuable mineral resources	
Favourable Factors (FF)	
Suitable features associated with characteristics of the	
– Geological formations	– Risk of natural disasters
– Hydraulic properties	– Procurement of land
– Geological environment	– Transportation infrastructure

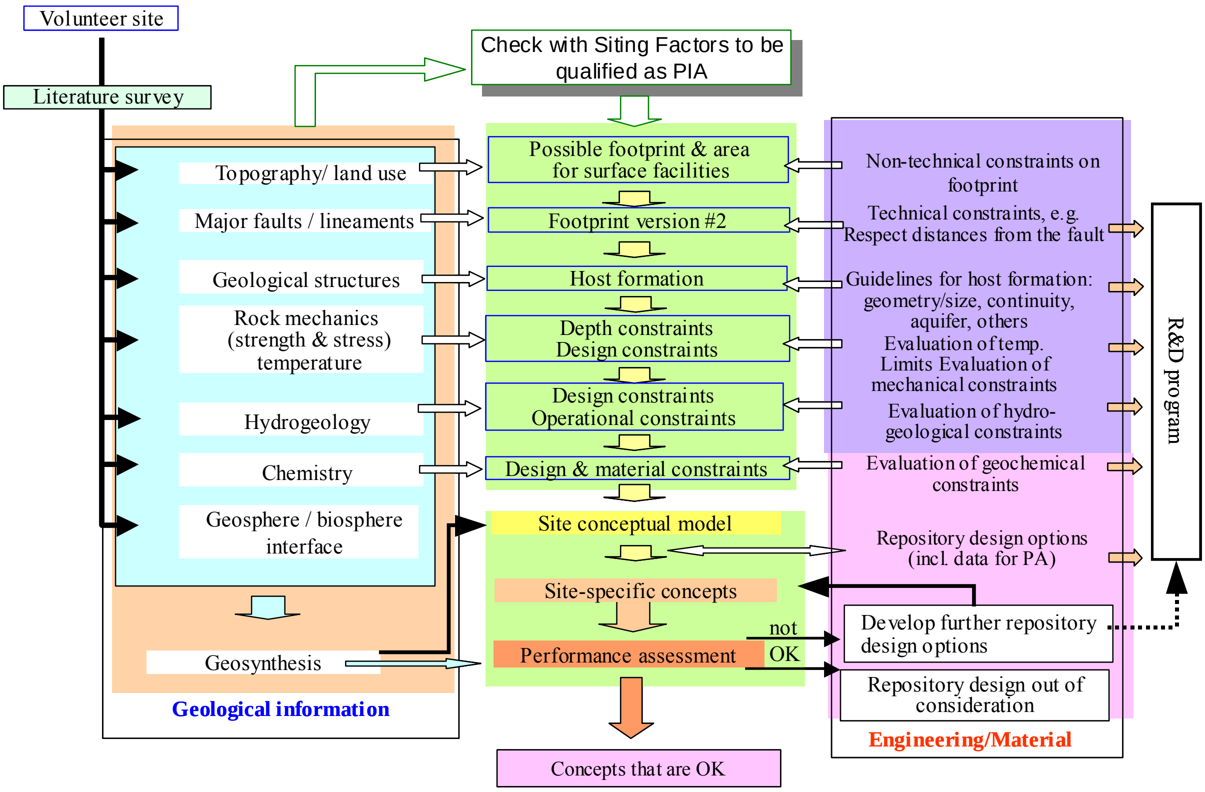
NUMO structured approach and general framework of repository concept development

A repository programme will extend over the best part of a century and this future period can be expected to involve major developments in science and technology and evolving socio-political conditions at regional, national and global scales, all of which could influence the repository programme. In order to maintain flexibility without losing focus and make the work more systematic, NUMO has developed a formalised tailoring procedure (e.g. Kitayama *et al*., 2005a;b), termed the NUMO Structured Approach (NSA). The NSA provides a methodology for developing repository concepts in an iterative manner, which couples management of immediate issues with consideration of longer-term developments. The NSA also guides the interaction of the key site characterisation, repository design and performance assessment groups and is facilitated by tools to help the decision-making associated with the tailoring process (e.g. a requirement management system, RMS) and with comparison of siting and design options (e.g. multi-attribute analysis). NUMO uses the general term "Repository Concept" to include repository design for a specific site environment, along with an associated description of construction, operation and closure, an assessment of operational and post-closure safety and an evaluation of socio-economic and environmental impacts (NUMO, 2004b). Therefore NUMO's safety case incorporates the framework of such repository concept development and emphasises the role of the NSA. Such an emphasis on the process of concept development rather than a single reference design is a direct consequence of the wide diversity of geological settings which might be found in qualified volunteer sites.

The NSA can be illustrated as shown in Figure 1, which shows the logical structuring of the output of the literature survey in order to narrow down the list of potentially appropriate repository design options. For example, the availability of sufficient space for disposal is a particular concern that arises from the small size of many Japanese municipalities, the need to keep respect distances from volcanoes and active faults, and the complex geological structures found in many regions of Japan. A first step is thus to determine if it is possible, at least in principle, to fit a repository into any potentially suitable formation. It is important to provide such a comprehensive and systematic logical structure for the development of repository concepts from the outset of the siting process, and then to apply it iteratively during later PI and DI stages, with appropriate modifications as required.

Initially, site characteristics may be rather poorly defined. Nevertheless, for any site which is not clearly excluded, all available information will be drawn together to outline site-specific repository concepts and to analyse these in a quantitative performance assessment (PA). Risk management for the repository concept development will consider uncertainty associated with the boundary conditions and input data for repository design.

Figure 1. **Block flow diagram for the development of repository concepts at the literature survey stage (based on NUMO, 2004b)**



H12 safety strategy/assessment basis and its extension by NUMO

The well-known H12 safety concept (JNC, 2000) provides the technical basis for NUMO's initial safety strategy and assessment basis at the early stage of the repository programme. However, there are extended requirements that go beyond the H12-type safety evaluation, resulting from the need to deal with a range of repository concepts and a variety of volunteer sites (NUMO, 2004b). The simplification required for early PA models was often so great that the analysis was completely insensitive to even rather major variations in site and repository concept properties. The simplified nature of the H12 PA models thus precluded comparison of the variants considered in the H12 project and the subsequent study by NUMO (NUMO, 2004b). At the various stages of siting and for repository concept development, PA models and process models should be as realistic as possible, so as to allow comparison and distinguishing between key features of different repository system options. In some cases, complex and heterogeneous geology may increase the relative importance of EBS performance to the safety case. Based on the H12 models and databases, further development has been discussed for the implementation stage of the disposal programme (NUMO, 2004b). Specific aims for such development include:

- More realistically representing the geometry of all components of the engineered barriers (essential for distinguishing between different repository design options);
- Including explicit representation of all materials present in the repository engineered structures and considering any significant interactions between them;
- Realistically representing the three-dimensional (3-D) geometry of the geosphere, with particular emphasis on the solute transport characteristics of all relevant formations;
- Developing a Japanese-specific biosphere model that contains the appropriate diet and lifestyle information and an improved representation of the geosphere/biosphere interface for both inland and coastal conditions;
- Incorporation of time dependency into the model chain in order to evaluate scenarios which evolve gradually with time;
- Improved assessment of uncertainties and their development in time and space;
- Increased efforts to test (verify and validate) models and databases.

These key issues have been further reviewed by consideration of specific model requirements at each site investigation stage. The focus for potential future developments has been discussed in terms of a 'wish list' for treatment of the EBS and the ambient geosphere (Ishiguro *et al.,* 2005). In this list, possible model development areas are identified for future repository design and development of a safety case, based on expected PA requirements at the literature survey, PI and DI stages. Major constraints on long-term model development are the extent of process understanding and the availability of databases. The list will be discussed further with modellers and data suppliers, to develop a plan for prioritisation. Nevertheless, model developments requiring a long development time should be started very soon. For example, NUMO has started the development of a 3-D mass transport model, which simulates radionuclide transport in a realistic 3-D geometry of repository engineered structures (Umeki *et al*., 2004).

Other supporting tools for management of safety case development

A Requirements Management System (RMS) is being developed to help implement the NSA (Kitayama, 2006). This RMS will allow the justifications, supporting arguments and knowledge base used for every decision to be clearly recorded and will highlight when such decisions may need to be revisited, for example due to changing boundary conditions or technical advances. It thus serves as a valuable tool to keep track of the wide range of constraints on designs (Sakabe *et al*, 2005), while the entire process runs within an overarching Quality Management System (QMS). NUMO has developed its own QMS to ensure high quality of all its technical activities, documents and databases. The QMS will be integrated within the RMS, to ensure the total quality of the repository project, including the safety case development. In the present absence of formal regulations on HLW disposal, NUMO has been establishing internal Working Standards (WS) for safety, that will become its code of conduct for early project phases and provide principles and guidelines on the basis of which the safety case is developed.

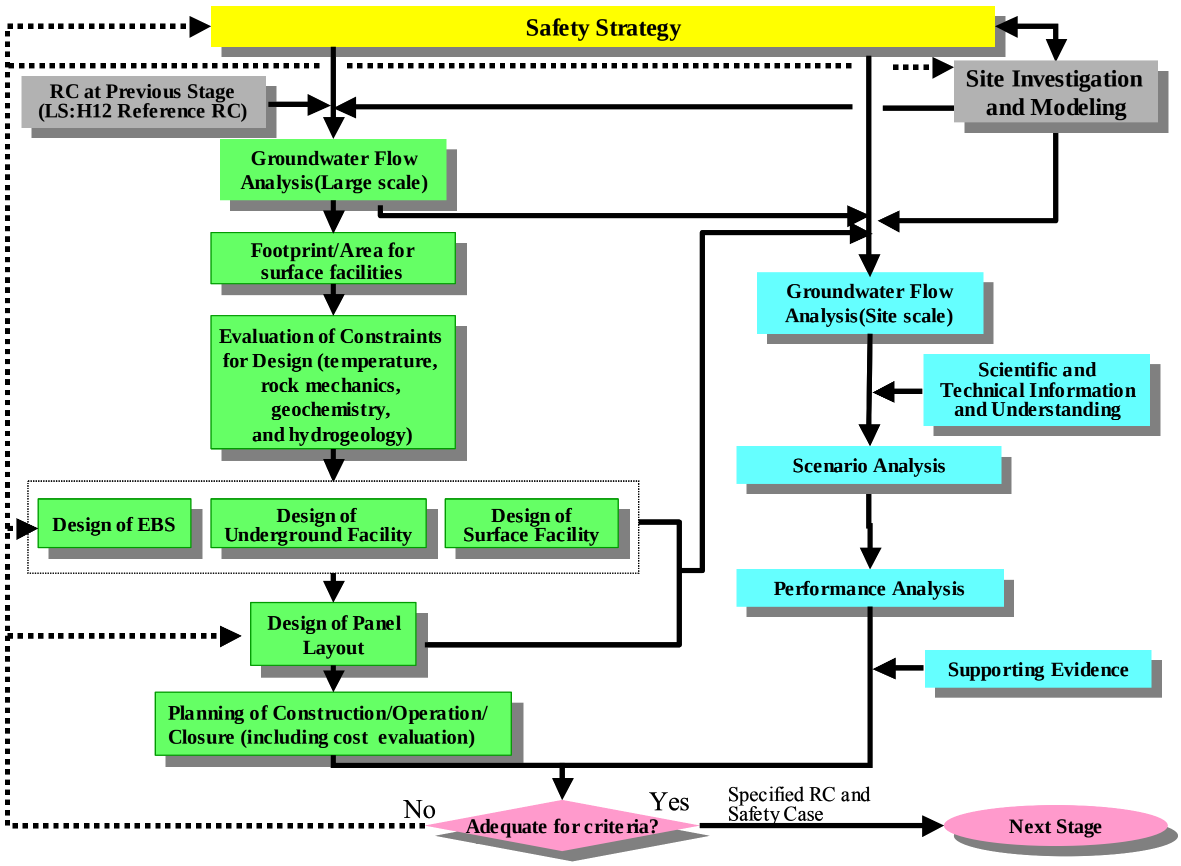
Stepwise development of safety case and its documentation

The safety case evolves with gradually increasing understanding of the host geological environment. Figure 1 provides key workflows for developing the repository concept for specific volunteer geological conditions. Figure 2, which focuses more on components associated with the safety case development, is a general workflow for repository design and PA at early stages of site investigations. An initial safety case can be established already at the literature survey (LS) stage, though less detailed technical evaluations and safety assessment will be involved, inevitably associated with quite a lot of uncertainties. The H12 safety concept and evaluations will provide a generic technical basis for the safety case at such early stages of the siting programme.

The aim of PA at the LS stage is to illustrate fundamental safety of the repository concept at the volunteer site, utilising evidence from the literature information for the site, complemented by generic and international experiences. Generally, available literature and site-specific database could be quite limited at this stage and the largest uncertainties may be associated with the geological environment. Little weight should then be placed on barrier performance of the geosphere at this stage, but EBS or near-field processes may be able to provide a robust safety case with minimal performance from the geosphere (predominantly isolation and protection of the EBS). Qualitative arguments in the safety case may be more meaningful than quantitative PA calculations at this stage, to scope uncertainties and identify data requirements for the PI programme and to provide strategy and guidance for the development of the RC and safety case at later stages. For the performance analyses at the LS stage, H12 PA models and support codes and generic databases may be sufficient.

At the PI stage, in which field investigations are initiated, more detailed technical evaluation is required within the safety case in order to justify the important (and politically sensitive) selection of DIAs. At this stage, the safety case will include more quantitative evaluations based on surface geological investigations and modelling, although availability of geological information will still be limited and significant uncertainties may remain. Site-specific data, based on several boreholes and geophysical investigations, will be available for the RC and safety case development, although again the importance of remaining uncertainties needs to be borne in mind. More detailed coupling between site investigation, repository system design and PA are critical for RC development at the PIAs, as indicated by the workflows shown in Figures 1 and 2. Models and codes need to be sufficiently detailed to take full advantage of likely information from surface-based investigations. An emphasis may still be placed on EBS performance in cases with limited geological information or complex and heterogeneous geology. A particular challenge is that a specific safety case is required for each PIA to support transparent comparison of different sites, locations and repository options. Development of a methodology for the comparison process (e.g. multi-attribute analysis) and associated assessment criteria will be critical issues, as will the implementation of comprehensive management tools (such as the RMS). The safety case at this stage also provides guidance for subsequent, more detailed investigations (including that in underground characterisation facilities at the DIAs) to reduce any identified uncertainties in the geological database.

Figure 2. **Workflow of repository design and performance assessment for the development of repository concepts and associated safety cases at early stages of site investigations**



The safety case at the DI stage will be required to be more convincing and complete to justify the construction of a repository at a selected DIA and to demonstrate compliance with regulations. Site-specific data from investigations in the underground experimental facility, with additional boreholes and detailed geophysical surveys, will provide critical input for the development of the RC and safety case at this stage. Models and codes need to be sufficiently detailed to take full advantage of likely information from the underground investigations. All relevant interactions between the EBS and natural barriers will be considered in design optimisation, while ensuring robustness of the safety case. All safety relevant information from diverse sources will be integrated into the safety case, to demonstrate compliance with regulations and form the basis for its presentation to other stakeholders. The RMS, coupled with an appropriate knowledge management tool, will be essential to support this task and ensure reliability of the safety case. The strategy for development of "next generation" models and codes as required for the PI and DI stages has been discussed elsewhere (e.g. Umeki *et al*., 2003, Ishiguro *et al*., 2005).

At the early stages of siting, major components of the safety case will be documented in the reports that are defined within NUMO's current repository programme. Site selection reports for PIAs, DIAs and a repository site are required by the Act. These reports describe the geological environment of the sites, results of the site selection process and relevant evidence/analyses/arguments for compliance with the siting factors and other legal requirements (e.g. regulation issues). These reports will be supplemented by reports on repository concept development, produced at the literature survey, PI and DI stages. These reports on RC development will include a site description, system designs, PA strategy, PA results and interpretations. Strategy and guidance to handling unsolved issues and uncertainties also will be documented in these reports.

Although development of the safety case at the early siting stages will be a challenge, key components will be documented by the regular publication of technical reports. Discussion of the specific documentation of an integrated safety case will continue, however, in particular aimed at the later stages of the programme.

Summary

NUMO's strategy for safety case development is constrained by the volunteering approach, along with initially short site selection stages. Developing the safety case will be a challenge as these constraints force NUMO to develop a tailored repository concept with associated safety case for each volunteer site within relatively short time periods and based on limited geological information. As such, linkages between site investigation, repository design and PA will be very important, to develop a safety case which maintains project flexibility without losing focus. The NSA guides such a process in a structured and iterative manner. NUMO's safety case is thus developed in the framework of the NSA. Another characteristic of the safety case development at early stages is a particular focus on long-term stability of geological environment coupled to detailed EBS/near-field analyses. Operational phase safety and the practicality of the engineering system will be also integrated into the total safety case. The safety case evolves with gradually increasing understanding of the host geological environment. In the early stage of the siting, key components of the safety case will be documented in the regular publication of the technical reports, such as site selection reports and reports on the repository concepts, that are defied in NUMO's current repository programme. However discussion of the specific documentation of the full safety case will continue, in particular for the later stages of the programme.

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THE CANADIAN NUCLEAR REGULATORY PROCESS AND USE OF THE SAFETY CASE FOR DEMONSTRATING THE LONG-TERM SAFETY OF RADIOACTIVE WASTE

P. Flavelle, M. Ben Belfadhel
Canadian Nuclear Safety Commission, Canada

The Canadian Nuclear Safety Commission (CNSC) regulates the use of nuclear energy and materials in Canada to protect health, safety, security and the environment and to respect Canada's international commitments on the peaceful use of nuclear energy. In this respect, the CNSC makes regulatory decisions on behalf of Canadian society. This paper describes the how the main elements of the safety case form an integral part of the licensing decisions made by the CNSC. The components of a safety case are a fundamental part of the information that the CNSC requires in support of a licence application or a pre-licensing environmental assessment. The use of the safety case is discussed in relation to the CNSC's open and transparent licensing process, where affected parties and members of the public are given the opportunity to be heard before the Commission through a public hearing process.

Introduction

Over the past several years, the regulation of the nuclear industry in Canada has evolved, achieving a milestone in 2000 with the coming into force of the Nuclear Safety and Control Act (NSCA) [1] and its related regulations. The NSCA is modern legislation that clearly establishes the regulatory authority of the CNSC with respect to the protection of people and the environment.

During this same time period the radioactive waste management community has developed the safety case as an approach to systematically structure and communicate an integrated collection of broadly-based safety arguments and evidence in support of project proposals. As early as 1986, the analytical components of the safety case were being discussed at an NEA workshop [2]. The concept of providing supportive statements of confidence in the results of a safety assessment was advanced in a 1999 NEA report by the *ad hoc* group on validation and confidence building [3]. This was followed by a 2004 NEA report [4] which further discussed the structure and content of a safety case, defined as "an integration of arguments and evidence that describe, quantify and substantiate the safety, and the level of confidence in the safety, of the radioactive waste disposal facility". The IAEA's coordinated research project on improvement of safety assessment methodologies for near surface disposal facilities (ISAM) further developed the methodological aspects of safety assessment, with particular emphasis on application to near surface facilities [5].

Since both the development of the safety case concept and the development of modern nuclear regulation in Canada reflect current practices in the nuclear industry, it is not surprising that there is significant consistency between the two. This paper provides an overview of regulatory requirements in Canada for demonstrating long term safety and examines the use of the safety case as a means to enhance that demonstration.

The Canadian Nuclear Safety Commission (CNSC)

Under the authority granted by the Nuclear Safety and Control Act and the regulations made pursuant to it, the CNSC is responsible to the people of Canada through Parliament for regulating the nuclear industry. Specifically, the mandate of the CNSC is: to regulate the development, production and use of nuclear energy and materials to prevent unreasonable risk to health, safety, security and the environment; to regulate the production, possession and use of nuclear substances, prescribed equipment and prescribed information; to implement measures respecting international commitments on the peaceful use of nuclear energy and substances; and to disseminate scientific, technical and regulatory information concerning CNSC activities.

The CNSC comprises a seven-member tribunal of Commissioners (the "Commission", who make independent licensing decisions) supported by a scientific and technical staff ("CNSC staff"). The CNSC's regulatory philosophy is based on two principles [6]:

- "Those persons and organisations subject to the *Nuclear Safety and Control Act (*NSCA) and regulations are directly responsible for managing regulated activities in a manner that protects health, safety, security and the environment, while respecting Canada's international obligations"; and
- "The CNSC is responsible to Canadians, through Parliament, for assuring that these responsibilities are properly discharged".

The CNSC's generally non-prescriptive approach to regulation places the burden of demonstrating safety on the licensee. The CNSC has developed a number of policies, guides and standards that provide direction and guidance to applicants on how to meet regulatory requirements. However, in most instances licensees are, free to use, with justification, the technology and methods they prefer to demonstrate compliance with the requirements.

The CNSC licensing process

The CNSC regulates all uses of nuclear energy and nuclear substances from cradle to grave (that is, from production to final disposition) through a comprehensive licensing process. The regulations pursuant to the *NSCA* can be categorised as those applicable to all licensed activities (for example the General Nuclear Safety and Control Regulations, the Radiation Protection Regulations and the Nuclear Security Regulations) and those specific for certain types of facilities (such as the Class I Nuclear Facilities Regulations, the Class II Nuclear Facilities and Prescribed Equipment Regulations and the Uranium Mines and Mills Regulations). These regulations specify the information that must be included in licence applications and set out the other regulatory requirements specific to the type of nuclear facility or activity involved.

For major nuclear facilities, the Commission, in accordance with its Rules of Procedure, typically holds public hearings over two days. On the first hearing day the applicant presents the application and CNSC staff presents recommendations concerning the application. The second hearing day, which is normally scheduled approximately 2 months after the first hearing day, is largely reserved for the Commission to hear interventions from the public and other stakeholders. After the conclusion of the hearing, the Commission deliberates *in camera* and then publishes its decision on the application with reasons.

However, in certain circumstances, before the Commission can exercise its authority under the NSCA, the CNSC, as a federal regulator, must comply with the Canadian Environmental Assessment Act (CEAA) and its regulations. An environmental assessment is a planning tool used by federal authorities to determine whether a proposed project is likely to cause significant adverse environmental effects before they exercise an authority that allows the project to proceed. Although there are different types of environmental assessment specified by the CEAA regulations (Screening, Comprehensive Study and Review Panel), they all provide for a thorough examination of the potential environmental effects (form both normal operations and from possible accidents and malfunctions), the available impact mitigation measures, and the significance of the residual effects. For the purposes of the CEAA, an environmental effect includes changes in the biophysical environment and the effects of those changes on socio-economic conditions. Public participation in the environmental assessment of a project is an integral part of the CEAA.

Only if an environmental assessment conducted pursuant to the CEAA concludes that the project is not likely to cause significant environmental effects, may it proceed with consideration of a licence application under the NSCA.

Regardless of whether the CEAA applies to a particular project proposal, an application for a licence must provide all information specified by the NSCA and its regulations, including among other things, detailed information on the proposed policies, programs and procedures for protecting the environment. CNSC staff performs an independent, technical review of the application and makes licensing recommendations to the Commission during the public hearing.

After the public hearing is completed, the Commission tribunal considers all of the information tabled and presented at the hearing by the applicant, CNSC staff and interveners, to determine whether "the applicant:

a) is qualified to carry on the activity that the licence will authorise; and
b) will, in carrying on that activity, make adequate provision for the protection of the environment, the health and safety of persons and the maintenance of national security and measures required to implement international obligations to which Canada has agreed." [7]

The Commission's decision and the reasons for the decision are published, normally concurrently with informing the applicant of the decision.

Licensing of long term radioactive waste management facilities

Most nuclear facilities require a number of licences over its lifecycle. For example, separate licences are required for site preparation, construction, operations, decommissioning and abandonment. This is a stepwise approach to regulation. At each stage of licensing, the elements, assumptions and (if appropriate) the licensee's performance are evaluated before the next stage is allowed to proceed. This results in a staged evolution of the demonstration of safety, where the content, detail and rigor of a safety case matures with each subsequent licensing stage (from environmental assessment through the licensing lifecycle to decommissioning and eventual abandonment).

This staged evolution approach applies to radioactive waste management facilities, which can range from mineralised waste rock piles and tailings impoundments at uranium mines, to engineered trenches for low-level radioactive waste, to dry storage facilities and geologic repositories for used nuclear fuel. Many waste management facilities are licensed as part of, a facility operating licence, while other waste management facilities are licensed separately. In Canada, most waste management facilities are for storage of the waste. Only a few types of facilities, such as uranium tailings management facilities, are recognised as disposal facilities.

The CNSC policies and guidelines concerning radioactive waste are written to be applicable to the wide range of facilities and activities that fall within its mandate. Regulatory policy P-290 [8] addresses the management of all types of radioactive waste. Regulatory guideline G-320 [9] deals with demonstrating the long term safety of managing any type of radioactive waste (since it is waste that requires long term management) based on the safety case paradigm.

Some of the regulations pursuant to the NSCA are applicable in common to all nuclear facilities (such as the General Nuclear Safety and Control Regulations [10] and the Radiation Protection Regulations [11]). Other regulations apply only to an identified type or class of nuclear facility. For clarity, subsequent discussions will be limited to the information required to be submitted in a licence application for a deep geological repositories for radioactive waste disposal, which would be subject to the Class I Nuclear Facilities Regulations [12] in addition to the regulations applied in common.

Pre-licensing environmental assessment

From a regulatory perspective, the first opportunity to develop a safety case is, as noted above, during an environmental assessment under the CEAA. As a planning tool, an environmental assessment for such a facility would normally include:

- a preliminary safety assessment to evaluate the environmental effects of the facility's construction, operation, decommissioning and abandonment;
- consideration of alternative means of implementing the project;
- development of a follow-up programme to address, if the project proceeds, the accuracy of the impact predictions and effectiveness of the selected mitigations measures; and
- public and other stakeholder consultation.

The information required to be presented in an environmental assessment is identified in project-specific assessment guidelines that are developed by the CNSC, in collaboration with other Federal and Provincial regulatory departments. These guidelines contain the regulators' expectations on how the range of potential environmental effects should be evaluated. For a waste disposal facility the project-specific guidelines are submitted for public comments before they are presented to the Commission for approval.

The environmental assessment describes the significance of residual (post-mitigation) environmental effects in terms of magnitude (or severity); duration (time period); frequency; reversibility; geographic extent; and ecological importance. Existing policies, standards and regulatory limits, as well as expert professional judgment are used in assessing significance. In general, at this early stage in the facility lifecycle there is limited information. The safety case prepared for an environmental assessment would normally be based on conceptual designs.

Application for a licence

Although the information required to be submitted in a licence application for a deep geological disposal facility varies between the types of licence needed at different stages in the facility lifecycle, the required information for each application has common elements such as:

- Detailed description of the facility and its operation (facility design and specifications for a construction licence, as-built facility and operations programmes and procedures for an operating licence, facility and programme performance history for renewal of a licence or for a decommissioning licence);

- Managerial information (organisational structure, qualifications of the applicant; quality assurance programmes; radiation protection programmes, environmental monitoring programmes, security, etc);
- Safety Analysis Reports (preliminary SAR based on the facility design for a construction licence, final SAR for operating licence based on the as-built facility and demonstrating the adequacy of the facility design, and updated SARs as appropriate based on operational performance for licence renewals);
- Environmental protection policies programmes and procedures, including any mitigation measures and follow-up programmes identified from an earlier environmental assessment;
- Decommissioning plans and financial guarantees; and
- Public information programmes.

The technical information (facility description, safety analysis, and environmental protection programme) is needed to identify, measure and mitigate the effects on the environment and the health and safety of persons that may result from the activities to be licensed at that particular stage in the disposal facility lifecycle. The quality of the technical information and the information on the applicant's organisational management and programmes reflect whether the applicant is qualified to carry out the licensed activities. The description of the facility operational procedures and programmes in the different safety areas inform the Commission whether the applicant will make adequate provision for the protection of the environment and health, safety and security of people.

Demonstrating long term safety of radioactive waste disposal

The regulations applicable to a deep geologic repository for radioactive waste disposal do not explicitly mention a safety case. However, the information needed to satisfy both technical and administrative aspects of the regulatory requirements for a licence application correspond to all of the elements of a safety case.

Regulatory policy P-290 on managing radioactive waste identifies CNSC staff expectations for the safe management of all types of radioactive waste, including operational management, long-term management and disposal. This policy includes the principle that "the assessment of future impacts of radioactive waste on the health and safety of persons and the environment encompasses the period of time when the maximum impact is predicted to occur". There is, then, no *a priori* time limit for demonstrating the long term safety of radioactive waste management, including disposal, and the maximum potential impact is expected to be assessed regardless of when it might occur.

CNSC regulatory guide G-320 discusses the development of a safety case in terms of performing a safety assessment and supporting it with complementary analyses and reasoned arguments. This guide draws upon focused guidance on safety assessments required under the Canadian Environmental Assessment Act, the NSCA and other Canadian legislation administered by government agencies having regulatory responsibilities (e.g. Environment Canada and Health Canada) and the Canadian Council of Ministers of the Environment. It also draws upon the developmental work undertaken by the NEA and the IAEA. The structure of the guide is consistent with the elements of a safety case as proposed by the NEA [4] and the methodology of a safety assessment as outlined by the IAEA [5].

The guidance on performing long-term assessments covers selection of a methodology, presentation of the assessment context, system description, time frame of the assessment, the assessment scenarios and the development and use of assessment models. The assessment context defines the purpose of the assessment (which establishes the level of detail in the assessment and provides justification for the assessment methodology) and demonstrates the applicant's understanding of the regulatory regime in which the facility will operate. The system description includes the site characteristics (including the environment) and the design and operation of the waste management system, at a level of detail appropriate for the purpose of the assessment and the stage in the licensing lifecycle of the facility.

The complementary analyses and arguments to support the safety assessment can be based on various combinations of different assessment strategies (such as performing scoping or bounding assessments, calculating realistic best estimates or conservative over-estimates, and using deterministic or probabilistic calculation techniques). Reasoned arguments can include assessments of natural analogues to illustrate the robustness of the waste management system and evaluation of complementary indicators of safety calculated in the safety assessment and complementary assessments.

Discussions

The *NSCA* and regulations do not make explicit reference to the term "safety case". However, all the elements that have been identified by the international community as important when assembling and communicating the safety case for long term safety of radioactive waste disposal facilities are embedded in the Canadian regulatory system. These elements include both technical and administrative/managerial aspects:

- Clear description of the system:

  The safety assessment at the heart of a safety case requires a clear description of the system being assessed: its site characteristics, facility design, operations and procedures, and decommissioning plans. The licence application also requires this information

- A safety assessment (depth and level of details vary with the licensing life cycle) to demonstrate the long-term safety:

  The safety assessment predicts environmental impacts. The licence application requires the environmental effects and impacts be shown to not result in an unreasonable risk to persons or the environment.

- Confidence building arguments based on multiple lines of reasoning:

  The safety case includes independent complementary assessments and reasoned arguments that are meant to enhance the confidence in the reliability of the results and conclusions drawn from the safety assessment. This helps provide confidence to the Commission and CNSC staff that the applicant is qualified and will provide adequate protection for people and the environment.

- Identification of uncertainties and follow-up programmes to resolve them:

  Specifically for environmental assessments, the assumptions, limitation and results of the safety assessment are analysed to identify outstanding issues that, along with verification of the assessment predictions where feasible, can be addressed through a follow-up programme integrated into the licence and hence subject to regulatory compliance verification.

- Stakeholder consultation:

  The stakeholder consultation process under *CEAA* and the public hearings format of the licensing process under the *NSCA* ensure significant public and stakeholder engagement, for which the safety case provides a useful communication structure.

- Commitment of the applicant to make adequate provisions for the long term protection of the environment and the health and safety of persons:

  The clear description of the facility design and operations allows the applicant to demonstrate that adequate provisions will be made to prevent unreasonable risk to the health, safety and security of persons and the environment, as required under the *NSCA*.

- Qualifications of the applicant:

  The safety assessment should clearly present the development of the scenarios in the assessment, the evaluation of the simplifications, assumptions and limitations in the assessment, and the interpretation of the assessment results. This contributes to demonstrating the technical competence of the applicant, which provides evidence to the Commission and to stakeholders that the applicant is qualified to carry out the licensed activities.

Concluding Remarks

In Canada, licensing decisions by the CNSC are informed by a wide diversity of information, beginning in the early conceptual assessment stages, through the design and operational stages. It is also significantly informed by public input. The regulatory regime incorporates a significant level of stakeholder consultation and interaction during pre-licensing environmental assessments and the public hearings licensing process. Even after licensing, the licensee is required to maintain an acceptable public information programme designed to keep those potentially affected informed about the actual and potential effects of the project on them and on the environment. Part of the mandate of the CNSC is also to disseminate scientific information on the uses and effects of nuclear energy.

The required information for licensing has evolved over time, and so too has the concept of the safety case been developed and elaborated. The safety case provides a useful framework for managing and communicating the safety-related information needed for a licensing decision. It also provides a disciplined structure for presenting and supporting information that exemplifies the applicant’s qualification, competency and commitment to safety of people and the environment - information that is needed by the Commission to inform its licensing decisions.

The safety case provides a disciplined approach to preparing and presenting the information required in a licence application. This can be a distinct advantage to both the applicant and the regulator. The safety case structure provides clear communication of the information to stakeholders and regulators, imparting credibility to the safety-related arguments. The configuration of a safety case also provides a template by which regulators can clearly communicate their expectations, and can be used to conduct consistent, structured regulatory reviews.

As use of a safety case grows and evolves, its usefulness for informing societal dialogue and licensing decisions can be expected to increase.

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THE ONDRAF/NIRAS SAFETY STRATEGY FOR THE DISPOSAL OF CATEGORY B&C WASTES

A. Dierckx[1] **W. Cool**[1]**, P. Lalieux**[1]**, P. De Preter**[1]**, P.Smith**[2]
[1]ONDRAF/NIRAS, Belgium; [2]SAM Ltd., UK

Introduction

ONDRAF/NIRAS is the Belgian Radioactive Waste Management Agency, responsible, among other things, for the development of a concept and design for the geological disposal of vitrified waste from reprocessing, spent fuel and long-lived low- and/or medium-level waste forms (termed B&C wastes in the Belgian Programme). The supporting science and technology programme is mainly focused on a reference argillaceous host formation (i.e. Boom Clay) and based on in situ data acquired in an underground research laboratory located in Mol/Dessel (NE Belgium), which is the reference site for methodological research.

ONDRAF/NIRAS has recently formulated a safety strategy for the disposal of B&C wastes that describes the process for developing (i), a concept and design for the geological disposal of these wastes and (ii), the evidence and arguments to show that these are both safe and feasible to implement as planned.

The safety strategy has been developed in the context of an iterative, stepwise approach to repository planning and implementation. Although the adoption of such an approach, and the specific decision points foreseen within it, are programme-planning issues that are not considered part of the safety strategy, the results of applying the safety strategy at any given programme stage provide key input to the decision-making process.

The same strategy is likely to be applied repeatedly at successive programme stages. However, the outcome, in terms of the description of concept and design, and in terms of the evidence, arguments, and analyses to support a safety and feasibility case1, will vary, as knowledge is acquired, boundary conditions and the requirements for decision-making change, and the assessment basis is refined. The end product of the process should be a concept and design for which the safety and feasibility case is adequate to support applications for relevant licences..

The documentation of the safety strategy, as presented in summary form in this paper, is intended to provide a firm basis for defending the concept and design currently under consideration, and to steer and prioritise R&D and demonstration activities towards underpinning statements regarding its safety and feasibility, called the safety and feasibility statements hereafter. The applicability of the safety strategy will be tested in the period leading up to up to the first Safety and Feasibility Case (SFC 1)

1. The term “safety case” is more widely used internationally (see, e.g. NEA 2004), although aspects of feasibility are generally included. By using the term “safety and feasibility case”, ONDRAF/NIRAS emphasises its view that ensuring feasibility (including operational safety) is as important a consideration as post-closure safety in developing a geological repository.

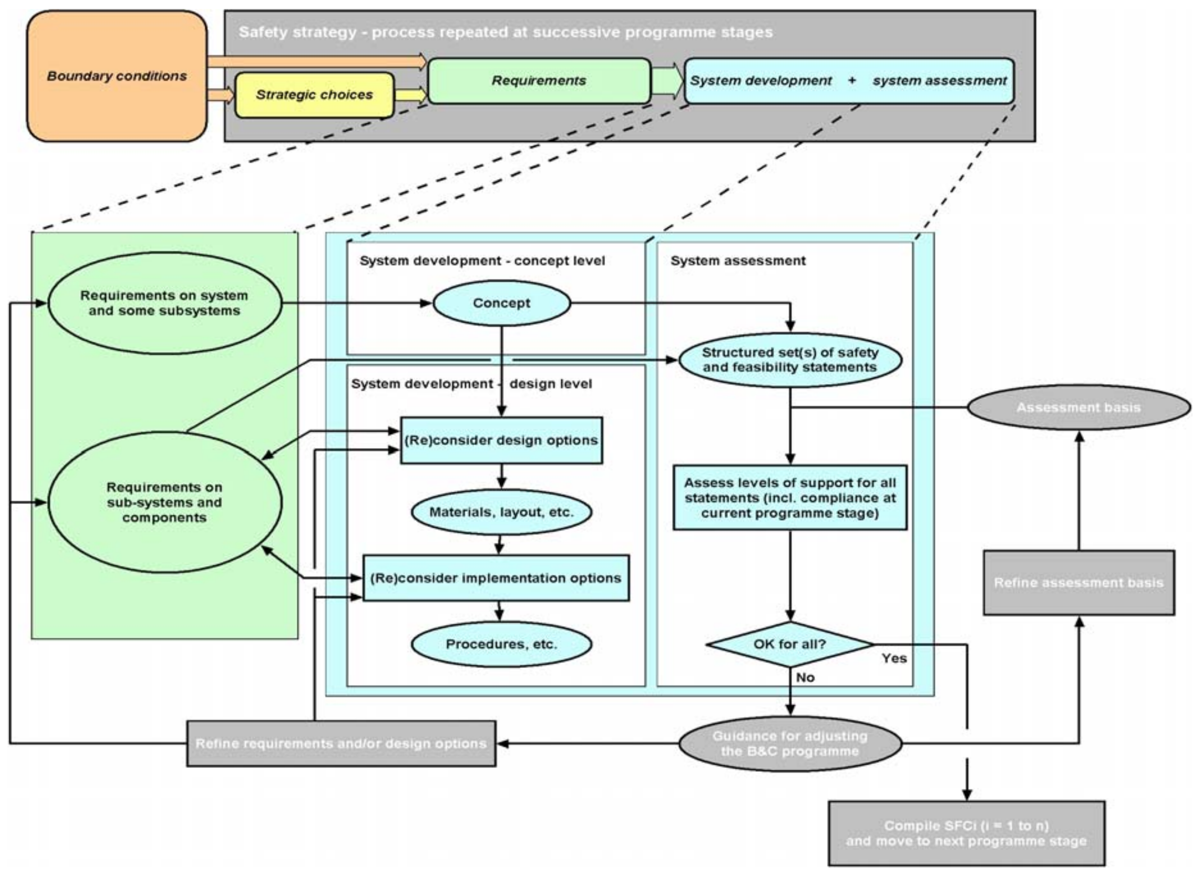
planned for 2013, the next scientific and technological milestone in the B&C Programme. SFC 1 will be based on the working hypothesis of disposal in Boom Clay, but will not be site-specific. However, together with other non-technical elements, including dialogue with stakeholders, it will form part of the information base that will guide any future siting decision. Should a decision be taken to proceed with siting, a second safety and feasibility case is foreseen. Based on the SFC 2, a go-ahead for a progressive industrial implementation could be given i.e. launching of the detailed engineering studies needed to prepare licensing applications.

Description of steps and outputs

Overview

An overview of the safety strategy is given in Figure 1. The safety strategy describes the process whereby, starting from a set of boundary conditions, a disposal system is developed and assessed, providing the necessary input to a safety and feasibility case.

Figure 1. **Overview of safety strategy, including major elements of the process (top diagram) and more detailed steps (rectangles – lower diagram) and outputs (ovals – lower diagram)**



Boundary conditions include, for example, international guidance and the Belgian regulatory framework and the "working hypothesis" that the Boom Clay will be the repository host rock for B&C wastes in Belgium. System development is guided by a number of strategic choices (consistent with the boundary conditions), which are translated into a set of requirements on the system as a whole, on sub-systems and on individual repository components. Consideration is given to what are the essential elements of the repository and its environment, and strategic choices are made, and requirements set, concerning what these elements will need to do, in broad terms, in order to meet the most fundamental objective of providing long-term passive safety.

System development

System development itself is carried out on two levels (see the lower diagram in Figure 1). At the higher, more general level is the development of a broad reference concept. At the lower, more detailed level is the making (or re-evaluation) of more specific design and repository-implementation choices.

The repository concept refers to broad system components and their functions, and not e.g. to specific materials and dimensions. It is developed based on the requirements set on the system and some sub-systems and is expected to be robust (i.e. insensitive) with respect to most reasonably foreseeable changes in *a priori* knowledge and boundary conditions. On the other hand, the detailed design process may well be modified as the programme progresses through successive stages, in order (i), to better adapt the repository to relevant boundary conditions and principles, (ii), to take advantage of advances in science and engineering and (iii), to enable better-substantiated safety and feasibility statements to be made.

Developing an engineering design is an iterative process. It involves:

- consideration of the design options available to realise the more generic concept; and
- consideration of the possibilities for implementing the design options.

Design development is guided by assessments of relevant factors, including long-term safety and factors related to feasibility, and the requirements placed on sub-systems and components. These detailed requirements are not, however, firmly fixed, and may be modified to some extent depending on the design and implementation options chosen (the two-way arrows in Figure 1).

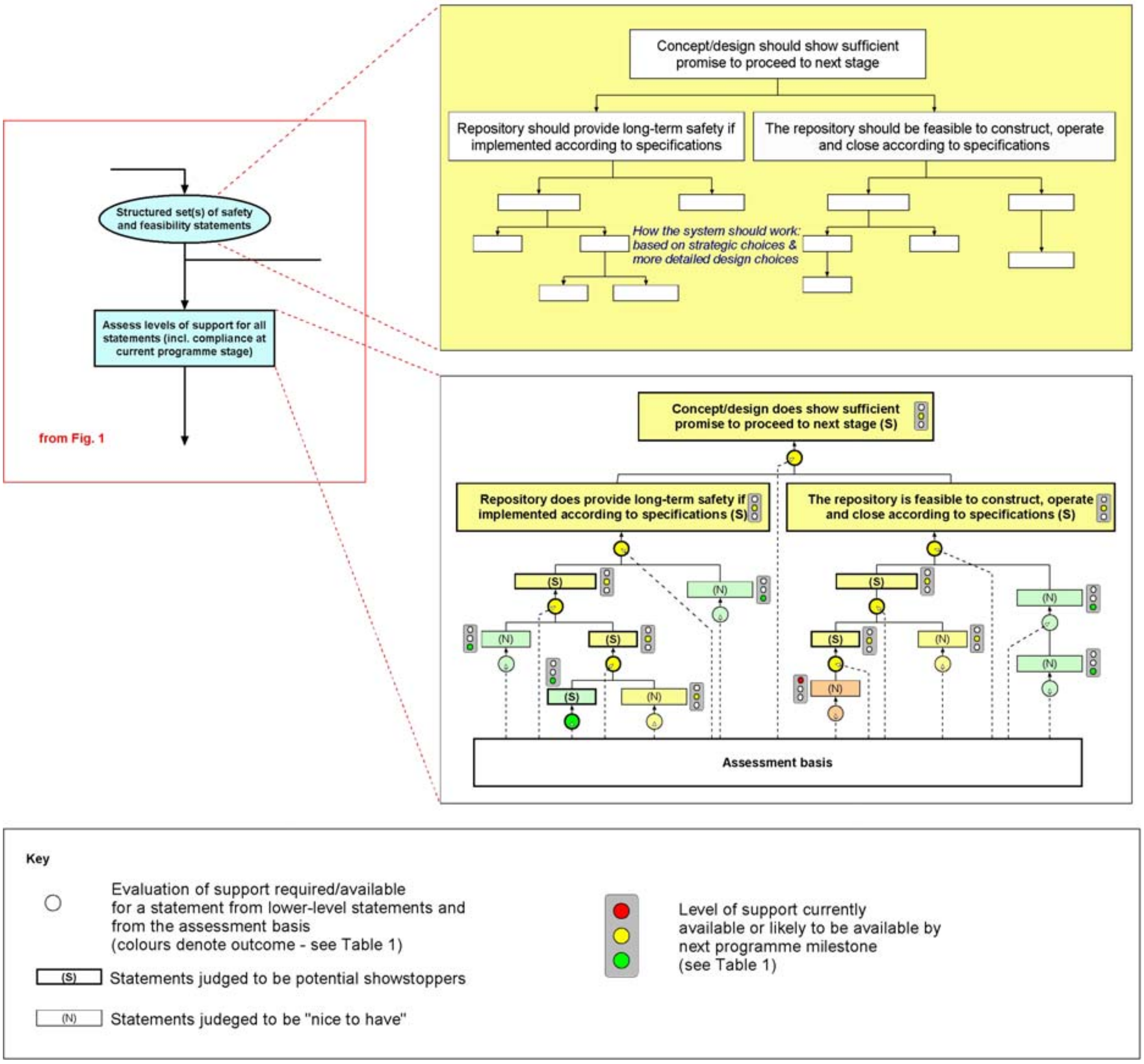
Development and assessment of a structured set of statements

Having selected a reference concept and design, these are evaluated in a process that involves:

- development of a structured set of statements regarding safety and feasibility; and
- evaluation of the levels of support for these safety and feasibility statements.

Safety and feasibility statements are developed and structured in a top-down manner, starting with the most general (highest-level) statements (e.g. the statement that the repository should or will provide long-term safety if implemented according to design specifications) and progressing to increasingly specific (lower-level) statements (e.g. statements regarding the evolution of specific system components). Lower-level statements are generally statements about what the system is designed or intended to do, or properties that it should have, in order to satisfy higher-level statements. This top-down structuring of statements is illustrated schematically in Figure 2 (top-right yellow box).

Figure 2. **Illustration of the top-down development (top yellow box) of a structured set of safety and feasibility statements (hypotheses) and the bottom-up evaluation (lower box) of the level of support required and available for each statement (claims).**



Safety and feasibility statements generally begin as hypotheses (i.e. statements of the type "the repository and/or its components **should** ..."), which may initially be tentative, and develop into increasingly well-substantiated claims (statements of the type "the repository and/or its components **will** ...") as the design and implementation procedures are developed and optimised, and the evidence, arguments and analyses that support a statement are acquired or progressively developed. In order to guide these developments, the support that is judged to be available and required for the various statements is evaluated. Specifically, safety and feasibility statements are evaluated according to:

I. How well supported a statement needs to be, or how critical the statement is, in the context of the current programme stage and of future stages?

II. What level of support is available, or is likely to become available according to the current planning of the RD&D Programme?

A classification scheme is proposed in Table 1 (although it has yet to be fully implemented and may be modified in the light of experience).

In order to evaluate general, high-level statements regarding safety and feasibility, more specific, lower-level statements that underpin them must first be evaluated. Thus evaluation of statements tends to be carried out from the bottom-up (Figure 2, bottom-right box).

Table 1. **Classification scheme for safety and feasibility statements**

		Type of statement	
		Nice-to-have" (N)	**Potential show stopper (S)**
Adequacy of the level of support judged to be available (or potentially available) with respect to the programme milestone at hand	**Changes to RD&D programme or design changes needed in order for adequate support to be achieved by next milestone**	**N1**	**S1**
	Good prospects for gaining adequate support with existing RD&D programme	**N2**	**S2**
	Adequate support judged to be available	**N3**	**S3**

This classification of statements provides guidance to the RD&D programme regarding the priority with which uncertainties or deficiencies in the assessment basis and in the design itself that need to be addressed. Particularly for the lower-level statements, the evaluation methods used may be informal in nature, and consist, for example, of qualitative expert judgement. In evaluating higher-level statements, on the other hand, a more systematic approach is generally adopted, including, for example, a feasibility assessment and a formal long-term safety assessment using a frozen assessment basis (6,7).

Application of the safety strategy

Strategic choices made in system development

As part of the application of the safety strategy, the following strategic choices have been made, constrained by current boundary conditions. Some of these translate directly into requirements on the concept and design.

1. In the case of heat-producing wastes, the engineered barriers shall be designed to provide complete containment of the waste and associated contaminants at least through the period when the heat output from the waste is high (no similar requirement for a containment is set for other wastes) to avoid the necessity to model contaminant transport during the thermal phase.
2. Given the "working hypothesis" that the Boom Clay will provide the host rock for a repository for B&C wastes, the repository shall be constructed at depth with this formation, with the overlying sedimentary formations providing the geological coverage.

3. Different categories of waste shall be disposed of in separate sections of the repository.
4. Repository construction and operation shall proceed as soon as reasonably possible, all necessary safety requirements and requirements related to societal acceptability having been fulfilled. This implies that one will start with the disposal of category B waste (LILW-LL), and afterwards continue with vitrified HLW and spent fuel.
5. The repository shall be closed (access routes backfilled and sealed) as soon as practically possible following emplacement of the waste.
6. There are preferences for permanent shielding, and for minimisation of operations in the underground.
7. There are preferences for materials and implementation procedures for which broad experience and knowledge already exists. The materials and implementation procedures shall not unduly perturb the safety functions of the Boom Clay, or any other component on which safety depends.
8. Repository planning shall assume that post-closure institutional control will continue for as long as reasonably possible, in order to reduce the likelihood of deliberate (unauthorised) or inadvertent human intrusion, taking into account the resources that this will require.

Reference concept and design

The current reference concept and design for a repository for B&C wastes is illustrated in Figure 3 (surface and subsurface repository facilities) and Figure 4 (supercontainer design for vitrified high level waste) and described in Bel *et al.* (1). The chain of activities needed to implement the current reference concept and design has not so far been planned in detail. Some tentative decisions regarding implementation have, however, already been made (consistent with the strategic choices outlines above), namely the decisions to opt for phased implementation and for permanent shielding.

Figure 3. **Surface and sub-surface repository facilities**

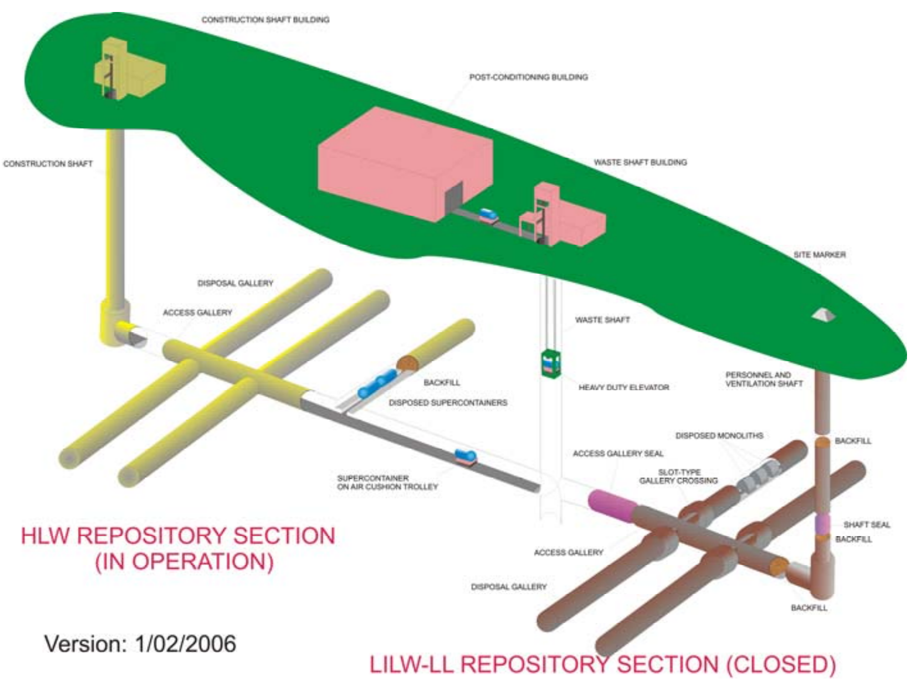
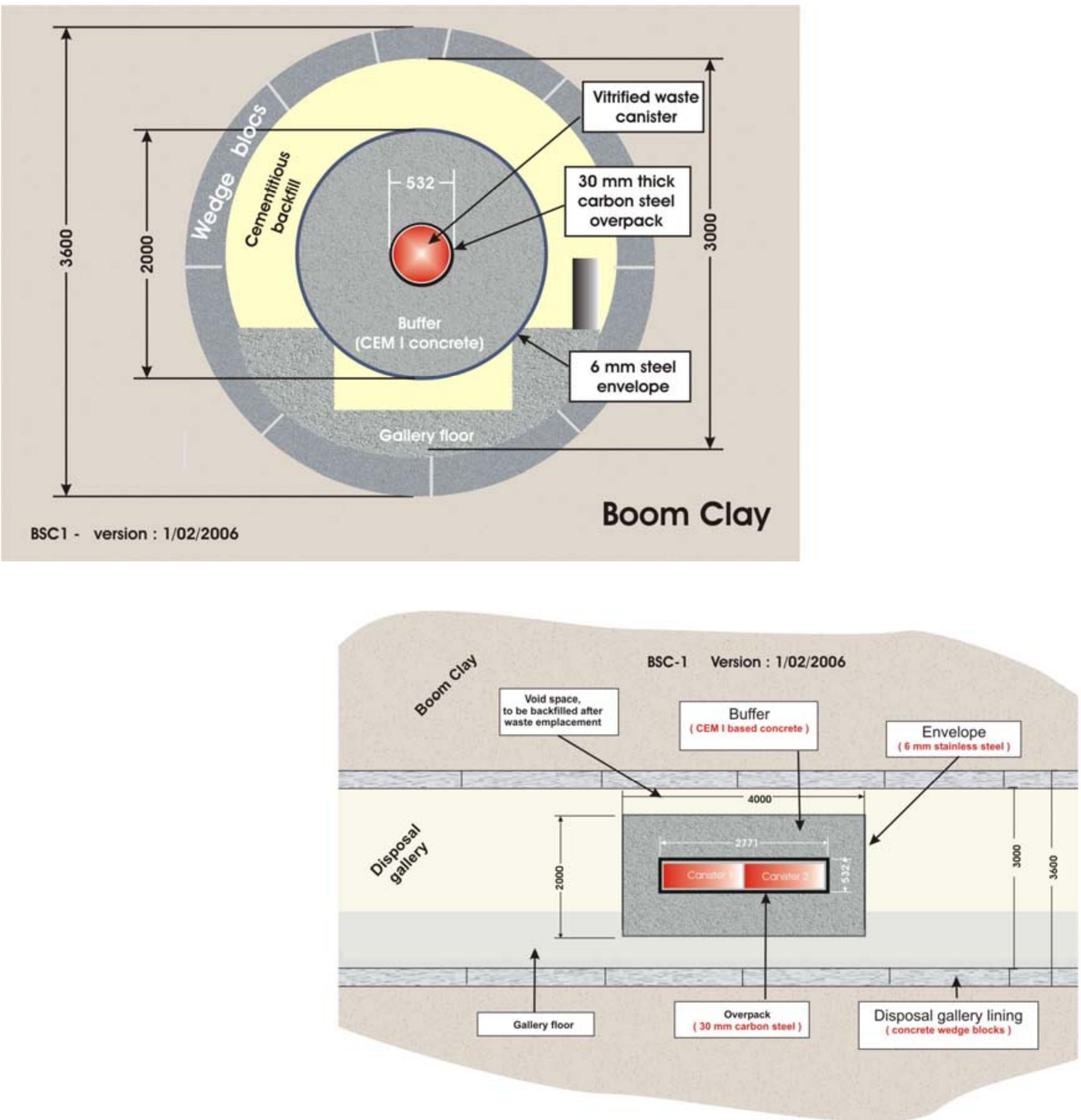


Figure 4. **Supercontainer for vitrified high-level waste within a repository disposal gallery**



Safety and feasibility statements and their assessment

In order that it meets the primary objective of providing passive long-term safety, any geological repository is required to:

- protect humans against contaminant release to the accessible environment;
- itself be adequately protected against disturbances by external events and processes; and
- protect humans from the possibility of exposure by inadvertent intrusion;

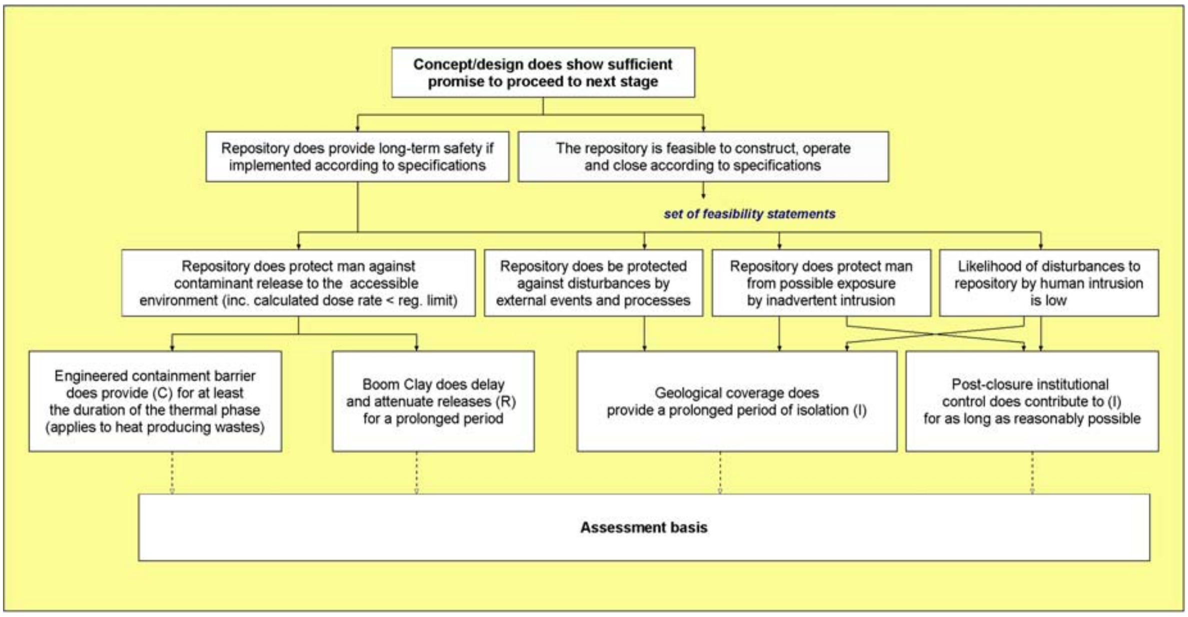
and to ensure that:

- the likelihood of disturbance to the repository by human intrusion should be low.

These generic requirements are satisfied by means of the long-term safety functions of isolation, engineered containment and delay and attenuation.

Figure 5 illustrates how such considerations are translated into the first few (generic) levels of a structured set of safety and feasibility statements. Lower-level statements are generally more concept and design specific.

Figure 5. Illustration of how long-term safety objectives are met, shown in terms of the top-down development of a structured set of safety statements



There is already substantial evidence underpinning some lower-level safety and feasibility statements for the current reference concept and design (below the levels illustrated in Figure 5), developed, for example, in the context of SAFIR 2 (2) (4), as well as relevant studies carried out internationally. A first, provisional assessment of the available support for the safety and feasibility statements has been carried out recently, to test the proposed safety strategy. The outcome will be presented at the symposium and will serve as a basis for the next RD&D plan.

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DEVELOPING AN ADVANCED SAFETY CONCEPT FOR AN HLW REPOSITORY IN SALT ROCK

J. Krone, N. Müller-Hoeppe
DBE TECHNOLOGY GmbH, Peine, Germany

W. Brewitz, J. Mönig
Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH, Braunschweig, Germany

M. Wallner, J.R. Weber
Bundesanstalt für Geowissenschaften und Rohstoffe (BGR), Hannover, Germany

Abstract

Preliminary Total Performance Assessments for HLW repositories in salt rock were performed within the framework of German R&D projects at the end of the 1980s and in the first half of the 1990s. In the meantime, several remarkable developments improved the basis for developing an advanced safety case:

- Surface and particularly underground exploration results of the Gorleben salt dome are available and provide an improved understanding of the geological system and corresponding processes.
- In the course of developing the closure concepts for the Morsleben Repository and Asse Mine, major progress was made regarding the design and performance proof of engineered barriers, in particular shaft and drift seals.
- Advanced tools have been developed and the knowledge base for assessing the consequences of radionuclide release in case of disturbed repository evolution has significantly improved.
- Contrary to the existing German safety criteria, the ongoing revision of these criteria intends to classify possible repository evolution incidents into scenarios with high and low probabilities.

In view of these changes, DBE TECHNOLOGY GmbH, BGR and GRS are developing and testing an advanced safety concept within a joint R&D project in order to identify the major needs for further R&D work. Whilst former safety assessment approaches for HLW repositories in salt rock used a conservative, even hypothetical, release scenario in order to show the compliance with dose constraints, the recent safety concept focuses on proving the safe enclosure without radionuclide release if the repository evolution remains undisturbed. Release scenarios are considered in the case of disturbed evolution only. This approach is considered to be more appropriate for a salt rock repository and takes advantage of its specific properties.

In this context, special attention is given to the proof of integrity (sufficient tightness) of the multi-barrier system, comprised mainly of the host rock as the main geological barrier and the engineered geotechnical barriers. The approach also focuses on appropriate scenario analyses that

allow the identification and assessment of remaining radionuclide release scenarios which cannot be ruled out.

Moreover, another emphasis of the project is to check in detail the physical and technical feasibility of all assumptions made for the safety assessment, since the technical feasibility is deemed an essential precondition for a safe repository not only in rock salt.

Introduction

In Germany, comprehensive F&E projects on the disposal of heat-generating waste and spent fuel elements in deep geological formations, mainly in salt rock, have been carried out in laboratory experiments and *in situ* for the past 40 years. Starting in the mid-1960s, fundamental and comprehensive *in situ* experiments were conducted in the exploration mine Asse. However, this mine is no longer available for research projects.

In connection with the suitability assessment of the Gorleben salt dome, a concept for the final repository Gorleben called "Update of the repository concept Gorleben" (Aktualisierung des Konzeptes Endlager Gorleben) [1] was compiled in 1998.

In addition to a description of the site, it contains a pre-conceptual planning of the repository based on the then expected waste volumes. The concept approaches all technical and safety-related aspects as well as issues concerning the licensing and the costs of the repository.

A comprehensive study of a repository for highly radioactive waste in salt formations consisting of the principal components:

- Characterisation and description of the geological formation
- Development of the technical design of a repository and assessment of its technical feasibility
- Operational evaluation and assessment of long-term safety

was carried out within the R&D projects:

- "Study on Alternative Disposal Techniques" (Systemstudie Andere Entsorgungstechniken – SAE (1984) [2]).
- "Analysis of Mixed Concepts" (Systemanalyse Mischkonzept – SAM (1989)[3]).
- "Analysis of Repository Concepts" (Systemanalyse Endlagerkonzepte – SEK (1996)[4]).

A common element of the long-term safety analyses carried out in this context is that they all emphasise the advantages of salt rock as the host rock for a repository on the one hand, i.e. its low permeability and its healing capacity which allow the safe isolation of the disposed waste. On the other hand, the compliance with protection objectives and regulations was proven with sufficiently conservative and thus comprehensive – but to a high degree hypothetical – release scenarios. Although these release scenarios are obviously related to disturbed repository evolution, they were assessed as though they represented the expected evolution of the repository.

During recent years a number of important research results and thus a substantial increase in knowledge were obtained which strongly affect the development of a reference concept for a final repository in salt rock as well as its safety assessment. The most important are:

- Detailed knowledge of the internal structure of salt domes in Northern Germany as a result of the underground exploration of the Gorleben salt dome.

- Development of engineering methods to prove the integrity of the salt barrier using dilatancy criteria as well as fluid and minimum stress criteria.
- Thermomechanical optimisation of repository design using validated constitutive models for salt and codes for 3-D modelling.
- Demonstration tests regarding the transport and handling of heavy loads with respect to the emplacement of casks and canisters containing spent fuel elements.
- Conclusion of the large-scale experiment TSS/Bambus to thermally simulate the disposal of containers containing fuel elements in drifts and to describe the compaction behaviour of crushed salt.
- Demonstration of the shaft sealing concept at shaft Salzdetfurth II.
- Development of concepts for the safety assessment of drift seals and concepts to control gas production in the course of the planning of the closure of the final repository ERAM.
- Studies on the optimisation of the "direct disposal" of spent fuel elements by emplacing fuel rod canisters in boreholes.
- Profound knowledge of the mobilisation and containment of radionuclides as well as on their transport in the surrounding areas and the geosphere.
- Further development of the tools to model and analyse radionuclide migration and their connection to thermodynamic data bases.
- Further development of the (computational) tools for the description of repository subsystems and integrated system assessment.

Furthermore, the safety criteria for the final disposal of radioactive waste have been refined taking into account international developments [5]. The recently established distinction between likely scenarios and scenarios with a low probability, and scenarios with a very low probability of occurrence is of special importance. For likely scenarios with a high probability of occurrence within the period of one million years, a dose constraint of 0.1 mSv is stipulated whilst the dose constraint for scenarios with a low probability of occurrence is 1 mSv and scenarios with very low probabilities of occurrence need not be considered during this period.

This distinction is of particular importance for the final disposal of radioactive waste in salt formations as in this case the aim is to take into account all likely events and to safely isolate the waste. Hence, the release of radionuclides needs to be considered for scenarios with a low or very low probability of occurrence only.

Taking into account these aspects, the proof of safe isolation of the disposed waste becomes vitally important for the proof of long-term safety of the repository as a whole. Complementary to this, radionuclide release is considered for those evolution scenarios where the safe isolation cannot be guaranteed.

Proof of Safe Enclosure

In an HLW repository in salt rock, the safe enclosure of the emplaced waste and the long-term containment of radionuclides should be ensured by a combination of geological and engineered barriers. Whilst the geological barrier is provided mainly by the tight and long-term stable salt as host rock formation the main function of the engineered barriers is to seal the artificial host rock penetrations which are bound to occur during the construction of a repository. The main engineered barriers are the shaft seals, the drift seals, the borehole plugs as well as the crushed salt used for the backfilling of the drifts and other underground cavities when closing the repository. In contrast to concepts for repositories in magmatic rocks, such as granite and tuff, and partially in clay, waste containers are not considered to be a long-term barrier for enclosing the waste and retaining radionuclides. Their function is to retain radionuclides until other engineered barriers become effective und safely isolate the waste.

In order to comply with the IAEA Safety Requirements "Geological Disposal of Radioactive Waste" [6], which require to reasonably limit the extent and probability of radionuclide release to the environment, an HLW repository in salt rock should be sited and designed in such a way that the probability of radioactive release is zero if the repository evolution is undisturbed. Thus, the safe isolation provided by the combination of geological and engineered geotechnical barriers should lead to a zero release scenario in case of undisturbed repository evolution. Therefore, the prevention of any brine inflow that could access the emplaced radioactive waste and thus form relevant pathways for radionuclide release is considered to be the main function of these barriers.

In order to prove the safe enclosure concept it will be necessary to prove the integrity of the geological and geotechnical barriers. Safe enclosure with zero release has generally been accepted as probable and undisturbed repository evolution although strong proofs supported by systematic proofs of the integrity of the geological and geotechnical barriers have not been provided in the past due to the missing scientific and technological basis. As a consequence, sufficiently conservative but hypothetical release scenarios have been used in order to demonstrate the compliance with regulatory limits.

Thanks to the scientific and technological progress summarised above, the proof of integrity for the main geologic and geotechnical barriers and correspondingly the proof of the safe enclosure of emplaced HLW and spent fuel in a repository in salt rock have become possible and provide the basis for the development of an advanced safety concept.

Integrity proof of the geological barrier

Former safety assessments for HLW repositories in salt rock considered the so-called "Anhydrite Scenario" as a representative reference scenario, which anticipated the existence and/or potential formation of pathways down to the emplacement level due to groundwater bearing layers in anhydrite banks bordering the rock salt formation that is used for waste emplacement. Recent investigations performed in the course of the Gorleben salt dome exploration programme showed that as a result of the salt dome evolution the anhydrite has been broken up into isolated blocks and thus, brine inflow and radionuclide release along the anhydrite banks can be excluded [7]. This finding questioned safety assessment approaches which use an "Anhydrite Scenario" as reference scenario, and was a major motivation for developing an advanced safety concept that is based on the integrity proof of the geological barrier comprised of salt as host rock.

Furthermore, dilatancy criteria have been developed [8] which allow the exclusion of the formation of any excavation damages or damages caused by mechanical impacts in those parts of the salt formation where mechanical stresses do not exceed dilatancy limits. Using such criteria and available gemechanical 3-D modelling tools, the repository structure can be designed in such a way that the tightness of the remaining salt barrier is not affected. In addition to this, the so-called brine pressure criterion should be considered as well. It ensures that saturated brines cannot penetrate the salt barrier if the maximum stress is not lower than the maximum hydraulic pressure resulting from the repository depths.

Furthermore, potential impacts, which may result from the geological evolution or from technogenic processes and could affect the salt barrier integrity, should be taken into account. Often discussed potential impacts are subrosion processes, thermal-induced stresses affecting the integrity of the top of the salt dome, movements of undetected brine inclusions, and hydrogen build-up pressures exceeding the fracture pressure.

Regarding subrosion of the salt barrier by unsaturated groundwater flows from the top of the salt dome, assessments of the Gorleben site showed that the extremely low average subrosion rates, low potential and velocity of the salt dome uplift cannot severely affect the salt barrier during the considered timeframe of 1 million years.

As regards the potential fracturing of the salt barrier due to hydrogen build-up it is important to note that hydrogen amounts and build-up rates strongly depend on the available amount of water. The water amounts available from dry rock salt are rather limited and can be managed safely, e.g. in special porous storages. Large amounts of gas which are able to build up critical pressures require sufficient amounts of free water. These scenarios should be disregarded for the proof of salt barrier integrity as part of the safe enclosure proof, because they can only occur if the integrity of another barrier is breached first. Thus, such scenarios are not crucial to the proof of safe enclosure as in fact they are not a cause but a result of the loss of safe enclosure. Generally, it is important to note that it is essential for the proof of barrier integrity and safe enclosure to address causalities correctly and to distinguish clearly between cause and effect.

Integrity proof of engineered geotechnical barriers

An advanced approach for proving the integrity of the engineered geotechnical barriers is addressed in sufficient detail in a separate paper [9]. Thus, major aspects that are important for the proof of the safe enclosure concept will be outlined here only.

To ensure the safe enclosure of the emplaced waste during the whole evidence period which is considered to be in the range of 1 million years, it is necessary to derive systematically the functional requirements for the geotechnical barriers, in particular for the shaft and drift seals which are to prevent any relevant brine inflow via the man-made penetrations of the salt barrier and any radionuclide release into the biosphere along these pathways. Major functional requirements are barrier tightness, characterised by the maximum acceptable permeability, and barrier durability, characterised by the necessary minimum performance period.

The development of the functional requirements for the engineered barriers is closely linked to the assessment of the compaction of crushed salt used as backfill material for access drifts and other underground cavities. The backfill compaction results from convergence which, on the one hand, is induced by the petrostatic pressure and, on the other hand, is accelerated by thermal expansion effects resulting from the heat released by the emplaced HLW and spent fuel. Whilst barrier permeability should be low enough to ensure a sufficiently slow saturation of the compacting backfill, the minimal performance period of the geotechnical barriers should exceed the time span necessary for the backfill to become sufficiently tight as a the result of compaction.

The choice of repository design, in particular the emplacement of HLW and spent fuel containers in drifts or boreholes, has major implications for the resulting functional requirements for the engineered barriers. The emplacement in boreholes has obvious advantages compared to the emplacement in drifts due to the significantly smaller amount of backfill and corresponding porous volume that needs to be compacted. Indicative calculations showed that crushed salt backfilled into boreholes with emplaced fuel rood containers will be compacted to nearly 100% within only a few years. In contrast to this, the results of the large scale heater test that was performed at the Asse mine to simulate the disposal of spent fuel containers in drifts, showed that – in this case – significantly longer timeframes are required until the crushed salt will be compacted adequately.

With regard to the feasibility of drift and shaft seals with a very low permeability and a long-term performance, major progress has been achieved during recent years as a result of the implementation of the closure concepts for both German LILW repositories in salt rock formations, the Asse mine and the Morsleben repository, and as a result of R&D works on sealing salt mines used as toxic waste dumps. Based on engineering designs and in-situ demonstrations, these efforts will lead to a systematic and comprehensive proof of full compliance with the functional requirements which are used in other parts of the safety assessments. Typically, it involves the proof of tightness or corresponding hydraulic resistance, proof of mechanical stability and limited crack evolution due to mechanical and thermal impacts, proof of long-term design and material durability and – last but not least – the proof of producibility, which should provide confidence that by using appropriate technological and quality insurance measures the geotechnical barriers built in-situ will comply with the performance requirements set out before.

Scenario development and assessment of release scenarios

Scenario development and analysis are commonly accepted tools for evaluating the variety of potential but not fully predictable repository system developments. As usual, the first step for their implementation is the compilation of a comprehensive FEP data base that includes any type of *feature*, *event* or *process* that could influence the development of the repository system. In the reported project, the NEA/FEP database was used as a starting point for such compilation. In a next iteration, the identified FEPs were evaluated regarding their relevance to the considered repository system that is defined by the geological reference model and the reference concept that corresponds with the repository design. Irrelevant FEPs, which are not expected to occur and influence the considered repository system during the considered evidence period in the range of 1 million years, were excluded. The remaining FEPs were specified in more detail. Special attention was given to a condensed description of the nature, the kind of influence on the repository system, the relevant timeframe of occurrence as well as to the correlation with other FEPs identifying the causalities between them.

Due to the huge amount of remaining FEPs and the large variety of identified correlations, scenario development is still considered to be an extremely complex task that is hard to manage. To reduce the complexity, the remaining FEPs were grouped into clusters according to their relevance to the repository subsystems (near field, host rock and far field) and according to their timeframe of occurrence (during initial temperature rise – first several hundred years, before the next ice age – first several tens of thousands years, remaining evidence period – up to one million years), where in each considered timeframe the relevant repository conditions are relatively stable. Following this approach, each cluster was filled with only a few tens of FEPs.

The next step in scenario development is based on the trivial fact that a further detailed investigation of scenarios which would not result into radionuclide release is superfluous. Thus, the scenario development should be focused on the identification of potential developments of the repository system which may lead to the formation of pathways for radionuclide release. Finally, screening of the remaining FEPs within each cluster should lead to the identification of those scenarios which are able to jeopardise the integrity of the geological and/or geotechnical barriers. In this context, the identification of these critical FEPs is directly linked with the integrity proofs of the geological and geotechnical barriers. Only those FEPs which cause impacts that may lead to the loss of barrier integrity should be considered as critical ones. They will provide the only starting points for developing corresponding release scenarios. To identify these critical FEPs, it is again important to consider the causalities between FEPs, e.g. FEPs which become critical only if the integrity of the barrier system has been affected before as a result of the occurrence of another FEP, should not be considered to be critical ones. Thus, a dry repository with an unaffected barrier system should be considered as the initial state for identifying critical FEPs and developing corresponding release scenarios. Despite this logic, former safety assessments used conservative release scenarios which were based mostly on postulated losses of barrier integrity neglecting the causalities which can lead or not lead to integrity loss.

It is expected that in a well-designed and well-sited HLW repository in a suitable salt rock formation, critical FEPs and corresponding release scenarios will only have a low probability of occurrence. In the recently proposed update of the German safety criteria, an upper dose constraint of 1 mSv has been introduced for such scenarios. For assessing such release scenarios, advanced modelling tools for source terms, radionuclide transport, and assessment of the radiological consequences are available and have been tested successfully. Moreover, modelling results for conservatively designed hypothetical release scenarios showed that these dose constraints can be met with significant safety margins.

Furthermore, it is expected that in a well-designed and well-sited HLW repository, FEPs which can affect the integrity of the geological barrier may be ruled out or might have at least a sufficiently low probability of occurrence. In that case, failure of the geotechnical barriers (shaft or drift seals) remains the only scenario that could jeopardise the safe isolation of the emplaced waste. Engineered barriers designed and constructed in compliance with good engineering practise and proven according to advanced safety concepts, such as the concept of partial safety coefficients, will have confidence levels in the range of 10^{-4} during their lifetime. Accordingly, the failure probability of such barriers will not exceed 10^{-4} during the performance period they are designed and proven for.

The reference repository design under consideration involves at least two independent sequentially allocated engineered geotechnical barriers, the shaft and drift seals. Thus, the malfunctioning of both barriers is a precondition for the occurrence of radionuclide release. If each of them has a confidence level of 10^{-4}, the probability that both will lose their capacity during their performance period does not exceed 10^{-8}. Consequently, in that case the occurrence probability of any release scenario is extremely low, i.e. less then 10^{-8}. According to the recently proposed update of the German safety criteria and in compliance with international practice such extremely low probabilities need not be considered in order to meet the safety criteria.

Even if the safety assessment were formally limited to the proof of safe enclosure by proving the integrity of geological and geotechnical engineered barriers it seems to be sensible to additionally assess a set of hypothetical release scenarios as so-called “what if scenarios” in order to demonstrate the robustness of the repository system. However, it is important to emphasise that these scenarios are just postulated, hypothetical ones and that their assessment is an add-on and does not belong to the proof that the safety criteria will be met.

Conclusion and outlook

DBE TECHNOLOGY GmbH, BGR and GRS are developing and testing the safety concept outlined above within a joint R&D project [10] funded by the Federal Ministry of Economics and Technology via its project unit at the Karlsruhe Research Center. The aim is to systematically and convincingly prove the safe enclosure and isolation of the emplaced HLW and spent fuel in case of a probably undisturbed repository evolution and consequently to prove zero release for that case. This approach does not only have obvious advantages regarding the communication of the associated safety concept to the public, but additionally benefits from and underpins fully the advantages associated with a well-sited and well-designed geologic disposal facility in a salt formation. Moreover, such a facility would fully comply with the requirement to reasonably limit the extent and probability of radionuclide release to the environment using design and siting criteria laid down in the IAEA Safety Requirements “Geological Disposal of Radioactive Waste” [6], and would not provide any reasonable leverage for further safety optimisation by considering alternative designs or sites.

The joint R&D project mentioned above does not have the objective to produce a safety case or a complete safety assessment for an HLW repository at a reference site. It is limited to the assessment of the feasibility to implement the selected safety approach in order to identify remaining shortfalls and deficits which may require further R&D efforts. Thus, it is considered to be a tool for focussing the German R&D programme on geologic HLW disposal on remaining priorities.

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THE PHENOMENOLOGICAL ANALYSIS OF REPOSITORY SITUATIONS (PARS) – APPLICATION WITHIN THE DOSSIER 2005 ARGILE (MEUSE/HAUTE-MARNE SITE)

F. Plas, P. Landais
Andra, France

Objective and definition of phenomenological analysis of repository situations (PARS)

Planning radioactive waste disposal in deep geological formations involves designing waste packages repository structures integrated into a natural geological environment to protect against the spread of radioactive elements contained in the waste. This requires an understanding of the interactions between the geological environment and the repository installations and an analysis of their development over time. Understanding the operation of a disposal facility thus means understanding the operation of a complex system subject to diverse and often linked phenomena and the need to take into account numerous scales of time and space. This knowledge contributes to the definition of both the safety *scenarii* and their treatments (dose calculations), and the irretrievability conditions.

The objective of PARS is to identify the phenomena acting on the evolution of a disposal facility. This evolution is that considered as the most probable according to the current scientific knowledge. In this evolution, each state of a disposal facility naturally depends on the previous state. For the sake of completeness, it is therefore necessary to analyse the development of the repository from the start of its construction up to times in keeping with the decay of radioactivity in the waste – approximately a million years (Figure 1). The evolution of the geosphere which forms the repository environment is also analysed in conjunction with the facility's size and its effects on this environment.

To analyse such a complex system it is advisable to break it into segments. A repository "situation" is one element in this segmentation. The repository segmentation retained is a time/space breakdown. Each situation corresponds to a phenomenological state of a part of the repository or its environment at a given moment in its life. For that purpose, the disposal and the geological medium are divided in components according to a tree structure. Thus the segmentation is made possible by the difference between the time characteristics of the different phenomena acting on the repository evolution, and is made easy by the repository's modular design. Thus it is possible to take advantage of some unsynchronised phenomena or the phenomenological independence between repository components to limit the modelling of a given "situation" to only those phenomena influencing repository evolution during this situation.

The repository situation phenomenological analysis input data are the repository designs under consideration and the data used to develop them, together with the body of available phenomenological knowledge.

Each repository situation analysis includes systematically:

- identification of thermal, hydraulic, mechanical, chemical and radiological phenomena to be taken into account, including an estimation of their magnitude based on current knowledge (Figure 1);
- description of the sequence of the phenomena ("process") and how they are linked together;
- identification of possible radionuclides release from waste packages or the paths used to transfer them into the repository and its environment;
- inventory of digital modelling relevant to the situation.

The analysis results are given in the "situation sheets". These are, for each repository situation:

- conceptual model of the situation state, specifying its characteristic dominant phenomena;
- conceptual models of the form of radionuclides release from waste packages or the paths used to transfer them into the repository and its environment;
- inventory of digital modelling means available;
- identification of design indecision or uncertainties linked to the state of phenomenological knowledge.

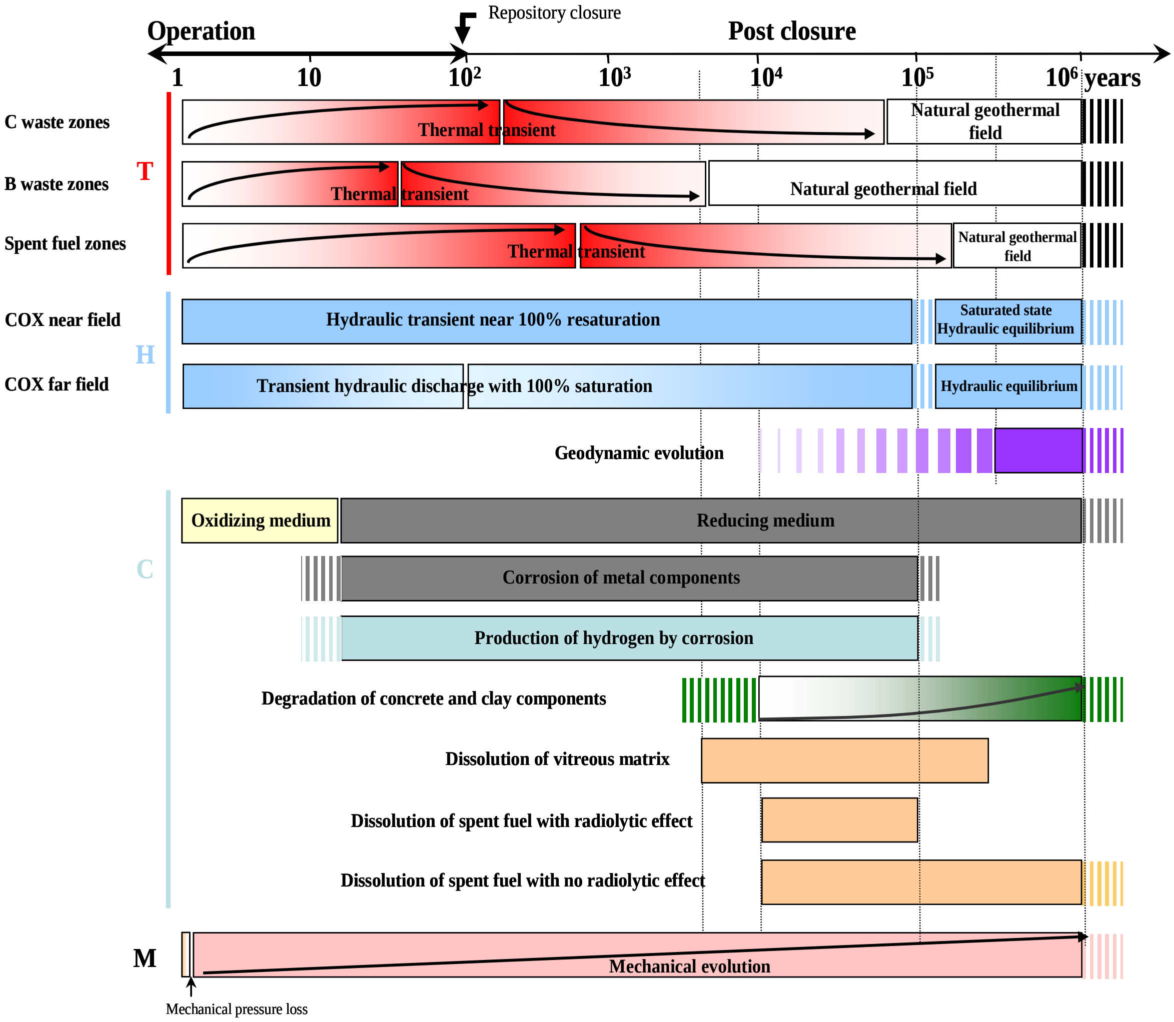
The conceptual models of repository situation states produce successive images specific to the life of a disposal facility. They are thus input data for digital modelling and simulations, particularly when describing the sequence and linking of phenomena specific to each situation.

They also act as a backbone in defining the monitoring programme for the disposal facility operation and surveillance/overview required to ensure that the process can be reversed. They assist in identifying phenomena to be observed and parameters to be measured as well as specifying the constraints linked to sensor installation.

The phenomenological analysis of different repository situations identifies links between design choice and phenomenology - the relationship between disposal solutions studied and the understanding acquired of their "behaviour" as regards waste containment, limiting radionuclides migration and delaying their possible transfer into the biosphere. It therefore acts as a system of reference for safety analyses, particularly in identifying indecision and uncertainties at analysis stage, and contributes to identifying fields for research.

It is also a system of reference in the design process: to clarify design option choices and adjust or modify the design to facilitate the understanding and modelling of repository behaviour. In particular, by identifying the successive physico-chemical states of a repository, its reversibility levels can be defined as the repository process progresses further.

Figure 1. **Chronogram of the major phenomena affecting the repository and its geological environment (based on the conventional hundred-year repository construction/operation/closure time diagram**



Input data

Geological media and waste inventory

The working hypotheses include in particular

- extent of knowledge acquired on the Meuse/Haute-Marne site. This geological data set includes mechanical, hydraulic, thermal, pore water chemical, solute transfer and retention properties.
- inventory and knowledge of radioactive waste (e.g. ILW, HLW and spent fuels, named respectively B, C and CU). The disposal vaults and the complete architecture sizing were based on the inventory model, particularly from the repository footprint point of view. Using this reference, the phenomenological analysis of repository situations clarifies the phenomenological consequences of waste inventory modifications specific to the repository situations being studied. The modifications concern the thermal characteristics of vitrified C waste and spent fuels in particular.

Repository design

The repository refer to disposal vaults for ILL waste (B waste), HLW (vitrified C waste), UOx/MOx spent fuels and repository facility architecture

Disposal vaults

Types of disposal vaults studied are:

- For ILLW: large diameter horizontal tunnels with an engineered barrier in concrete.
- For HLW: short horizontal tunnels without engineered barriers (reference) short horizontal tunnels with an engineered barrier in clay. The vitrified waste package over-pack is in carbon steel.
- For Spent Fuels: horizontal tunnels with clay engineered barrier and steel over-pack.

Repository architecture

As far as phenomenology is concerned, design choices aim at simplifying phenomena which could influence the development of a repository and thus their understanding and modelling, in particular:

- For the repository general architecture, disposal areas for ILLW, HLW, UOx Spent Fuels and MOx Spent Fuels are separate and arranged on one level in the middle of the Callovo-Oxfordian clay layer: this facilitates modelling and safety analyses and furthermore makes modifying the inventory and the repository process itself more flexible.
- From the hydraulic point of view, the general disposal architecture, but also that of the disposal vaults, is a “cul de sac” (dead end) design in an attempt to limit, “a priori”, hydraulic connections inside the repository. Modelling is made easier when the choice is made early on. In return, the construction and operation of the repository can give rise to constraints requiring re-examination in view of the safety analysis results (operational and long-term safety).

From the thermal point of view, temperature criteria are used as the basis for sizing designs for disposal of vitrified C waste and UOx and MOx spent fuels (number of packages per vault, distance between vaults, type of canister, etc.). Thermal criteria imply that temperature should never exceed 90°C in the host rock and that temperature should be lower that 70°C after 1 000 years. Designs are sized so that the leaching of the vitrified C waste by the Callovo-Oxfordian water does not start before the temperature has dropped below 50°C, as the leaching rate increases with the temperature.

Furthermore, current understanding of the chemical behaviour of radionuclides means that modelling their transportation is more uncertain for temperatures above 80°C. In the early stages, before safety analyses, preliminary repository designs avoid such configurations, which make the phenomenological analysis and modelling easier. On the other hand, this implies design constraints for spent fuels in terms of disposal volumes or canister specifications.

In view of the separation of disposal areas, underground structures are broken down at a second level into disposal areas for B waste, C waste, UOx and MOx spent fuels, service and connecting drifts between these areas and the shafts, and the shafts themselves as a whole.

The underground structures are listed on several levels in the tree diagram. In particular, disposal areas distinguish between disposal modules, then disposal vaults and vaults components (package, engineered barrier, vault plug, etc.), then specific aspects of the vault components (swelling clay of a vault plug, etc.). Not all repository components at the tree diagram lower levels are defined at the preliminary design stage. The analysis identifies definition needs and introduces hypotheses which are clarified. For example, the water, electrical and communication networks required by the repository or monitoring operations are not defined. It is assumed at this stage that they have no influence on the phenomenological evolution of a repository.

In this analysis, waste packages are considered as components within the repository. Their behaviour before emplacement in the repository is taken into account as input data, but is not included in this analysis.

The geological environment is divided by geological formation: the “Callovo-Oxfordian clay” repository host formation, the Dogger “water-bearing” carbonate formation beneath the argillite which is not intersected by any repository structure, the “water-bearing” Oxfordian limestone above the argillite, the Kimmeridgien marl and near the surface the Tithonian limestone, all three being intersected by the repository shafts. This section of the tree diagram concerns the geological environment in its natural state, “undisturbed” by the repository. The geological environment sections disturbed by the repository, in particular the repository vaults walls (“near field”), are attached to the repository area tree diagram (“disturbed geological medium (MG) area”).

Disposal process

General repository architectures studied are based on the assumption that repository operations will be carried out over a hundred years or so. In particular, this takes into account interim storage times of 30 to 60 years for vitrified C waste and spent fuels to meet repository thermal design criteria.

To facilitate the phenomenological analysis of a repository during the different phases of its development, repository operations have been phased on the hypothetical basis of a simple and progressive sequence. There is continuity in operation and a regular, average rate of disposal in keeping with the waste arrival input.

A repository theoretical procedure model has been established as a hypothesis for the phenomenological analysis of repository situations during its operation. It acts as a reference for analysing the phenomenological influences of the different disposal modules between themselves, particularly from the thermal point of view.

Following the inventory model, the time chart depicts B waste disposal over 60 years, C waste disposal over approximately 50 years and UOx and MOx spent fuel disposal over approximately 40 years. By considering the interim storage and cooling (through ventilation) times, the operating life of the repository is extended to around 100 years.

Definition of repository situations

The breakdown of the phenomenological development of a repository into “situations” is a time/space breakdown. Breaking down a repository system in space is based on the first levels of the tree diagram. It is justified by the repository’s modular design. Breaking down the repository system in time distinguishes between the periods before (pre-closure) and after (post-closure) the repository is closed, in other words during and after the first hundred years of a repository evolution.

The repository's operating period

The time breakdown is based on a succession of operations carried out in the repository. These operations trigger permanent or temporary changes in the state of the structures: digging of vaults & cavities, fitting them out and ventilation naturally trigger modifications in the mechanical, hydric and chemical status of the Callovo-Oxfordian argillite; emplacement of waste packages modifies the thermal and radiological status of the facility: sealing the structures triggers the start of their re-saturation and a provisional chemical status.

For the disposal areas, the evolution of a module determined by these operations can thus be described according to the following situation sequence:

- before the structures are constructed, a Callovo-Oxfordian area is designated to become a module;
- digging and fitting out the module and its vaults;
- waste package emplacement;
- pre-sealing phase before the disposal vaults are sealed;
- sealing the vaults;
- closing the module handling and internal drifts;
- pre-closing phase before the connection drifts and shafts are closed;
- closing connection drifts and shafts.

The duration of each of these phases is specific to each of the B waste, C waste and UOx and MOx spent fuels disposal areas.

The phenomenology linked to excavating a module varies with its situation in the facility architecture. To take account of this, situations describe clearly the phenomenology of a module j distinct from module i in the reference analysed. The aim of this distinction is to assess design flexibility towards modifications in the disposal process.

The period after closure of the repository

The time breakdown is based on its phenomenological evolution: in fact, there are no further operations likely to modify the facility evolution. As long as the geological and geo-dynamic context remains the same, the evolution of the repository is basically determined by its own phenomenology. The different thermal, hydraulic, mechanical, chemical and radiological phenomena have different time characteristics which determine the successive, distinctive states of the facility. It is possible therefore to define a "typical sequence" of situations by distinguishing between:

- a thermal phase for waste disposal areas giving off heat;
- a re-saturation phase for excavated structures, once closed. This re-saturation defines new physical states for the facility: it triggers chemical material exchanges, particularly between the disposal vaults and the surrounding geological environment. It takes into account the hydrogen production phase related to the corrosion of steel packs and overpacks as well as the radiolytic processes;

- a phase during which the mechanical evolution of the structures is determined by progressive damage to their supporting structures and coatings;
- with the damage to supporting structures and coatings and the evolution of other components in the different modules and drifts (back-filling, sealing, vacuums, etc.), the repository is subject to the generalised mechanical load by the geological environment.
- in the very long term, when the mechanics have stabilised, only material exchanges survive, particularly between the repository zones and the surrounding geological environment.

How long these different situations last depends greatly on the nature of the structures. In particular, the thermal phases have very different time spans for C waste (a few hundred years) and spent fuels (thousands of years). In the same way, the dimension of the tunnels for C waste results in quicker re-saturation than the B waste ventilated large tunnels.

The geological environment determines the context of repository evolution. The initial state of the geological state is modified in the long term by forecast climatic changes on the scale of several tens of thousands of years (ice age). This justifies distinguishing a situation for the most superficial or "water-bearing" formations through possible modifications of the hydro-geological flux linked to these climatic changes.

In the shorter term, the effects of natural erosion of surface formations must also be analysed.

In the very long term, the aim of the geological environment analysis is to identify the elements of a reasonable forecast of its geo-dynamic evolution and the consequences on hydraulic regimes.

For the surface installations, two situations are differentiated before and after the repository closure. The aim is to identify possible links between the surface installations and the evolution of a deep disposal facility, as for example keeping the excavated material (argillite, limestone) for back-filling purposes.

The analysis of the evolution in the surface environment is also based on two situations: before and after closure. In the long term, this evolution analysis is carried out with that of the geological environment.

The phenomenological analysis of radionuclides leakage and transfers in the repository and its environment forms an essential input to the safety analyses by the Phenomenological Analysis of Repository Situations. However, considering the concentrations and masses in question, the presence of dispersed radionuclides after release does not influence the phenomenological evolution of the repository and its environment. Consequently, the time/space breakdown of the facility evolution analysis is not based on the conceptual evolution of a mixture of radionuclides.

Situation sheets

The definition of situations constitutes the reference framework for the phenomenological analysis of a repository. Analysis sheets specific to each situation are produced covering the following points:

- A first part defines the situation from the temporal and material point of view. When does the situation analysed start and finish? Which repository parts and components are concerned? Furthermore the hypotheses underlying the situation analysis are described: hypotheses linked to preliminary designs or those specific to the phenomenological analysis.

- The second part describes the phenomena acting on this situation. For the sake of completeness, the analysis examines systematically all the thermal, hydraulic, mechanical, chemical and radiological phenomena likely to influence the facility evolution. The link between design choices and phenomena to be taken into account is analysed for each situation. The surviving indecision and uncertainties are identified. As they are required for safety analyses, the possibility and methods of radionuclides leaking and being transferred are investigated and described for each situation.

- The third part summarises the phenomenological analysis in the form of conceptual situation models. On the basis of this analysis, the conceptual model describes the sequence of phenomena ("process"), their respective presence and how they are linked together. In particular, the dominant or "triggering" phenomenon determining specifically the physico-chemical state of the facility during the situation analysed is specified.

- The fourth part is dedicated to the digital modelling of the situation: once the processes described in the conceptual models have been mapped out, an inventory of existing digital models and simulations applicable to the modelling of the analysed situation is produced. This part constitutes an input into the digital modelling studies ("digital modelling platform").

KNOWLEDGE MANAGEMENT: THE CORNERSTONE OF A 21ST CENTURY SAFETY CASE

H. Umeki, H. Osawa, M. Naito, K. Nakano, H. Makino
Japan Atomic Energy Agency (JAEA), Japan

I.G. Mc Kinley
McKinley Consulting, Switzerland

Abstract

The safety case for a radioactive waste repository involves many complex, multi-disciplinary issues; these must be summarised in a comprehensive and concise manner, with links to all supporting information. The safety case can thus be considered an edifice built on structured knowledge. Knowledge is defined here in the very widest sense; including all of the information underpinning a repository project. Knowledge management covers all aspects of the development, integration, quality assurance, communication and maintenance/archiving of such knowledge. When seen from this perspective, the exponential expansion of the knowledge base represents a little-discussed challenge to safety case development. Indeed, knowledge production rates in this area are rapidly reaching, if not already surpassing, the limits of traditional management methods.

This problem has been recognised in Japan and thus a project has been initiated to develop a "next generation" knowledge management system (KMS). This will utilise advanced electronic information management technology to handle the vast quantity of material involved. Autonomic systems will perform many of the information processing functions, helping ensure that required knowledge is accessible to all stakeholders and that gaps can be identified and supporting R&D prioritised. In a departure from conventional structuring by technical discipline, the prototype KMS utilises a safety case structure. This should facilitate use of the core of "neutral" scientific and technical knowledge by both the implementer and the regulator. Flexibility is built into the system, to allow it to be restructured to match the user's needs or even interfaced directly to a formal requirements management system.

Introduction

There has been considerable effort over the last few years aimed at developing a standardised terminology and a logical flow for the processes associated with the safety assessment of geological disposal of radioactive waste – in particular the development of a "safety case"[1,2]. This complements efforts over several decades to accumulate the fundamental understanding, databases and modelling tools needed to carry out quantitative assessments. One area that has received little attention, however, is the formal management of all of the diverse information that supports the safety case – predominantly because, until recently, this could be handled rather informally. Although the knowledge base was large and rather multidisciplinary, it was not beyond the capabilities of widely experienced generalists to have a fairly comprehensive overview of all relevant work within a national programme – or even an international overview of particular disposal areas. This situation has, however, changed considerably as a result of the exponential growth of the knowledge which needs to be considered for a safety case as it is now defined. The need for formal knowledge management methods is now critical in all major programmes – whether it is recognised or not!

Here, the term "knowledge" is taken to be a global term, which encompasses all of the science and technology (implicitly including social science, economics, medicine, etc.) which underpins a repository project. This can be classified as common knowledge (e.g. that established in component disciplines – e.g. geology, chemistry, materials science, civil engineering, etc.), generic waste management knowledge and project-specific knowledge. Knowledge is not static, but evolves with time in line with the general progress in science and technology.

The term "knowledge management" as used here covers all aspects of the development, integration, quality assurance, communication and maintenance/archiving of knowledge – including data, understanding and experience. It is an active process, which is focused by specific programme or project requirements (themselves developed and structured by a requirements management system (RMS) [3]. Ideally, knowledge should be objective and value-free but, in practice, it is inevitably conditioned by the opinions of the expert staff involved and their cultural environment – particularly in areas that are novel or involve interactions between several technical disciplines. An important aspect of knowledge management, therefore, involves the evaluation of potential biases, in addition to more standard assessment of conceptual and data uncertainties. Furthermore, experience is associated directly with individual staff and also accumulates as their career progresses. Therefore, an active programme of experience transfer is needed to ensure that it is passed to younger generations before older staff members retire.

Knowledge Management is a term commonly used in many areas of technology but, in general, focus is on conventional approaches to systematic handling of technical information [4]. In the 1980s, or even in the 1990s, it was possible for top managers in the nuclear waste field to have a reasonably comprehensive overview of all relevant technical work contributing to a repository project. Since then, there has not only been breathtaking growth in basic knowledge, but work has become more international and has been opened up to wider scrutiny – with increasing emphasis on non-technical aspects associated with open communication and public acceptance.

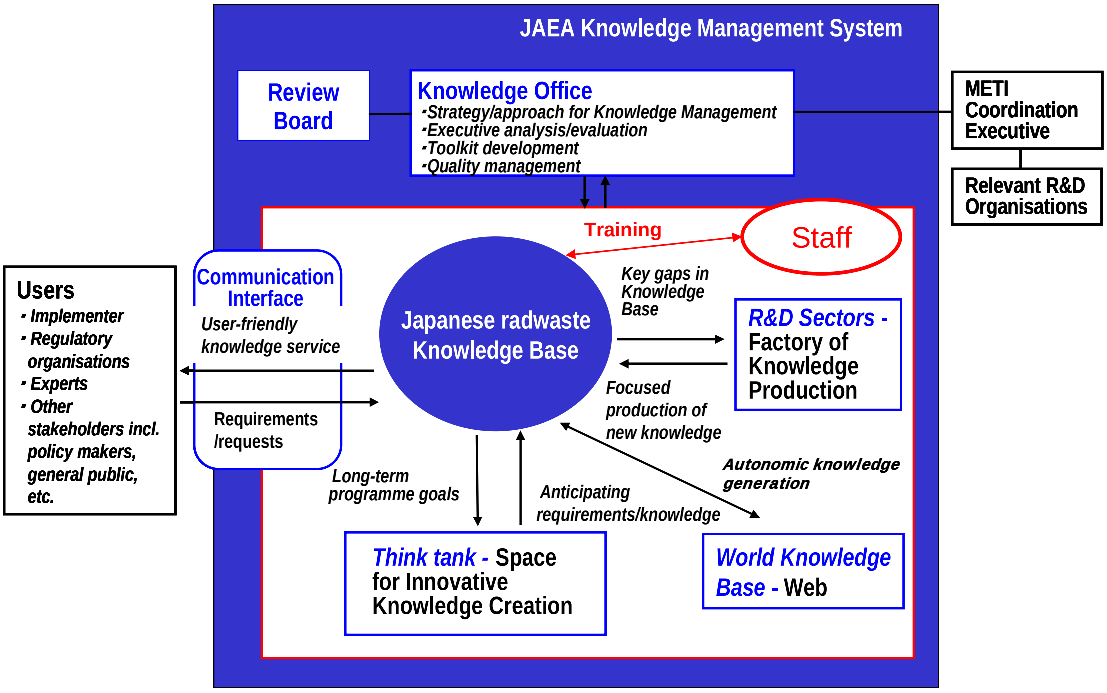
Except possibly in the smallest and most isolated programmes, the traditional approach is beginning to lead to an obvious loss of overview, synthesis and flexibility. In retrospect, some of the problems experienced by the very technocratic approach to repository programme development can be attributed to the narrow range of knowledge considered to be significant by key decision makers (decide – announce – defend approach leading to safe concepts which were unacceptable to stakeholders). More recently, the move towards devolving much decision making to consensus amongst stakeholders may lead to the opposite situation – compromise concepts which are not straightforward to assess and/or tricky to implement.

The problems resulting from the sheer volume, range and production rate of knowledge can be largely attributed to advances in technology and, thus, it appears that they can be overcome only by making best use of the most advanced technology available. Luckily, the problem is not unique to radioactive waste management and hence there are many analogous areas where existing experience can be utilised. Nevertheless, special challenges arise because of the long timescale of radioactive waste disposal projects – licensing of repositories possibly occurring towards the middle of this century, based on safety cases which are already under development at present. This means allowance must be made for huge changes in the boundary conditions under which the case will be presented.

The knowledge management concept

The basic concept for the knowledge management system (KMS) has been developed taking account of the fact that safety cases will be a major input for stepwise decision making in the Japanese HLW repository project [3,5]. Figure 1 illustrates the structure and major elements of the system being considered.

Figure 1. **The structure and elements of the knowledge management system**



The essence of the concept is that all relevant information will be held in a central electronic database (backed-up appropriately). Already, most key documentation is produced in electronic form and associated raw data and all manipulation and modelling of it is available on digital files. Key supporting references from past work can be scanned in and any critical data from earlier work can be digitalised. The greatest challenge is format standardisation and ensuring that information remains readable, despite evolution of software and hardware in future decades. These problems are, however, common to large-scale projects that are now ongoing in many different fields to produce electronic archives. The archive must be fully searchable by methods allowing contextual analysis of written text and autonomic analysis of data. This is certainly not a trivial job, but requires processing capacities which are minor compared to some of the massive and complex data mining projects being developed, and implemented, by national security agencies [6].

A central Japanese nuclear waste management database is necessary, but not sufficient, for the envisaged applications however. This needs to be complemented by access to the Internet, which allows focused data mining of both material produced by other national waste management programmes, and also relevant information which lies outwith the formal radioactive waste area – e.g. fundamental developments in geology, materials science, mathematical modelling, etc.

From this starting point, specific knowledge management applications can be developed. Initially, for example, these could focus on support of the production and review of publications. Review, in particular, has suffered considerably due to the volume of material being produced – even prominent technical journals have been forced in recent years to retract high profile papers due to failure to identify fundamental problems that, in retrospect, were rather obvious. To improve this situation, the KMS would allow a draft document to be analysed; for example including:

- Autonomic review checks – spelling (including technical terminology), grammar, equations and data manipulation (including reporting and propagation of uncertainties).
- Identification of previous publications covering the same area – checking completeness of referencing.
- Consistencies/inconsistencies with other relevant material (linguistic and technical):
  - Text duplication (also identifying plagiarism).
  - Similarities/differences in data handling or interpretation.
  - General compatibility with documented consensus.

After analysis, documents can be archived with hyperlinks to all review analysis and referenced material. Although certainly not replacing peer review by qualified experts, such analysis would be better than the many cases where no review at all occurs for low level documentation and would assist in the normal peer review process, if this is carried out. The peer review itself, plus any response to it, would also be hyperlinked to the finalised document and would be a resource for future autonomic reviews.

Clearly this document analysis function takes a lot of effort to establish and, initially, will need extensive expert checks. Automatic learning systems (e.g. expert systems or neural networks) can, however, build on this effort to ensure that, with time, it become more powerful and needs less direct supervision.

A further application based on the same basic analytical software would involve improved communication and access to the knowledge resources. As the knowledge base expands and becomes increasingly interlinked, information searches can become increasingly complex, based on natural language description of the material that is desired. Output could be envisaged in the form of a customised, automatically generated report with hyperlinks to relevant supporting information.

In terms of communication, it is recognised that users will include not only technical specialists, but also lay stakeholder groups – including politicians and the general public. To facilitate communicating relevant information (combating the acknowledged problem of "knowledge asymmetry"), appropriate output interfaces will be developed, which emphasise simple natural language (at the level of technical blogs) and presentation of data in the form of autonomically generated visual material, including animations. Indeed, this could probably be extended to automatic chat-room management for special groups/topics (e.g. site characterisation activities at a particular location). Although some questions may need to be forwarded to experts, it is likely that a large percentage of interactive communication could be handled by smart software.

The proposed structure allows gaps in the knowledge base to be identified and provides guidance for the R&D needed to fill them. The main limitation is that this tends to be constrained by expectations based on the current technical state-of-the-art. For projects running over many decades, the consequences of future advances in science and technology should also be considered. Although there are clearly limitations to the extent to which this can be done, the approach of using focused "think tanks" will be investigated as a complement to more traditional methods of setting R&D goals and priorities (see Figure 1).

Knowledge base development and structuring

The knowledge base represents the fundamental inventory of information that is needed to meet the requirements of users. Guidance for developing and structuring the knowledge base comes predominantly from studies of the evolution of the Internet. Even though it is acknowledged to be difficult to predict Internet development over even the next two decades [7], some trends seem clear. Of relevance here is the increased departure from traditional structuring of databases and archives, which focus on grouping information by technical discipline. This can be replaced by flexible structuring focused on specific applications, with hyperlinks crossing disciplines as appropriate. Ideally, this can be directly interfaced to the RMS of users – if they adopt such a structured approach of defining their information needs [8]. Initially, however, in the absence of such RMS, a structure is being established based on a generic safety case [1] for HLW disposal in Japan [9] as indicated in Figure 2.

As indicated, the requirements for producing and communicating the safety case can be organised in a top-down hierarchical manner, which then results in a bottom-up flow of knowledge, which is manipulated to produce increasingly generalised statements which are part of the "case" that a repository at specific location is safe – given the caveats set by the uncertainties inherent to early stages of a stepwise implementation programme. In line with the previous definition of knowledge, the components of this knowledge base are listed in Table 1. The table also shows the content of particular knowledge components and indicates some of the key management processes involved. Table 1 highlights the sub-components which can, at least to some extent (except "experienced synthesis team" and "experienced coordination team"), be facilitated by electronic tools and databases.

Table 1. **Components of JAEA knowledge base**

Form of knowledge	**Management functions**	**Content**	**Required developments**	**Comments**
Data	Data management	**- raw data (internal)** **- solicited data (external)** **- processed data**	- autonomic QA - internal & external data mining - autonomic data processing	Potential area for international collaboration
Documents	Document management	**- internal documents** **- key external documents**	- robust archive - autonomic QA / cataloguing / cross-referencing	Electronic archiving critical problem area
Software	Software management	**- archive of all relevant codes / databases** **- archive of manuals & handbooks** **- archive of relevant output**	- robust archive - autonomic change management - formal approaches for QA	Electronic archiving critical problem area
Experience & methodology	Resource management	**- procedure manuals & guidebooks** **- expert systems** **- training materials**	- use of expert systems to preserve experience - training approach for the next generation	Much of requirements could be addressed by national (regional?) training centre
Synthesis	Knowledge integration	- experienced synthesis team **- expert systems**	- description of key integration processes - approach to QA	Needs considerable development to automate
Guidance	Knowledge coordination	- experienced coordination team	- prediction of requirements (Think tank) - process for filling key gaps in knowledge	Very difficult to automate
Presentation	User / producer dialogue	**- user friendly interfaces (interactive – allowing dialogue)**	- high-end graphical methods for presenting complex information	Should be tailored to needs of different stakeholders

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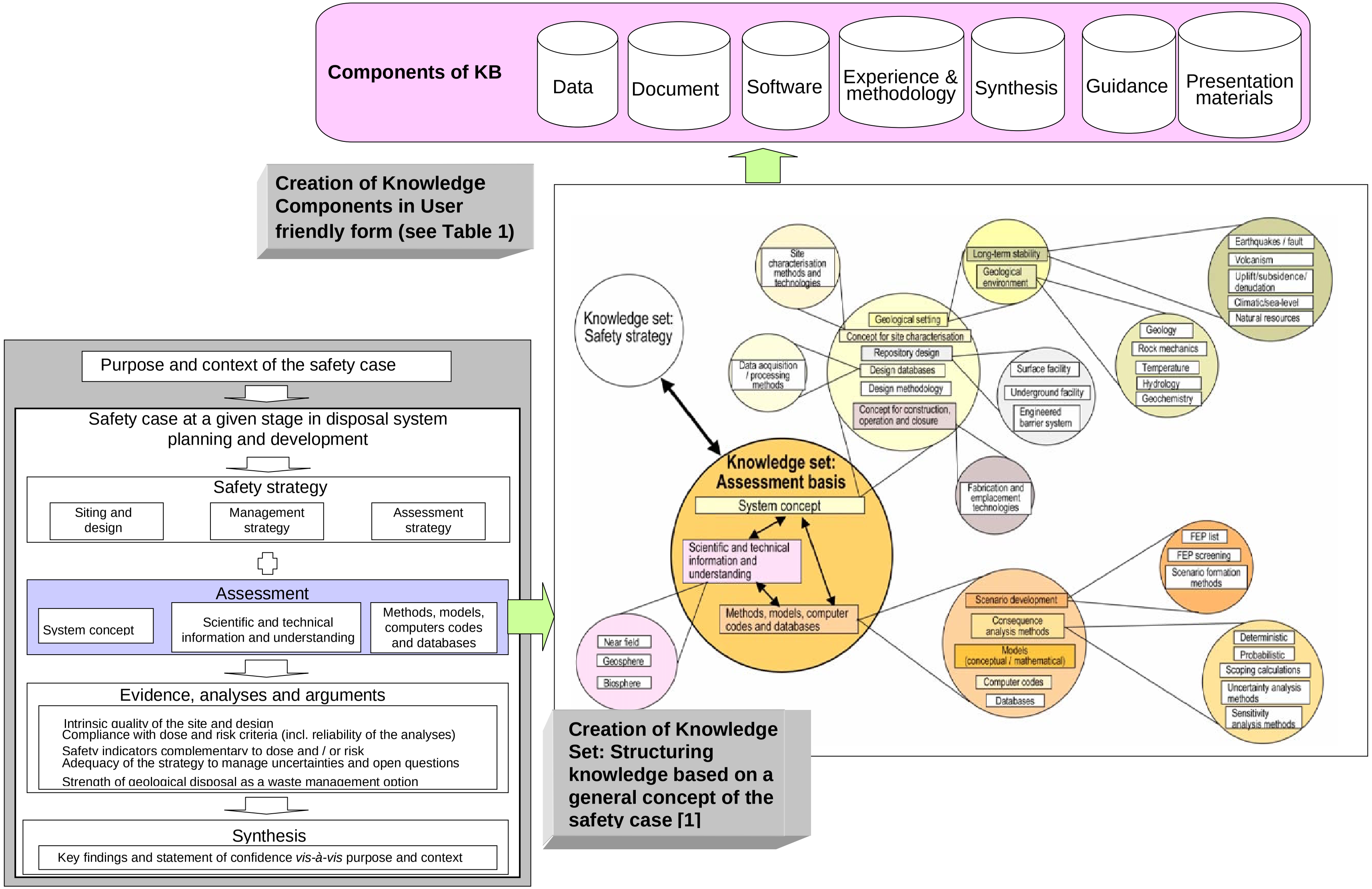
Figure 2. **The contents and structure of knowledge**

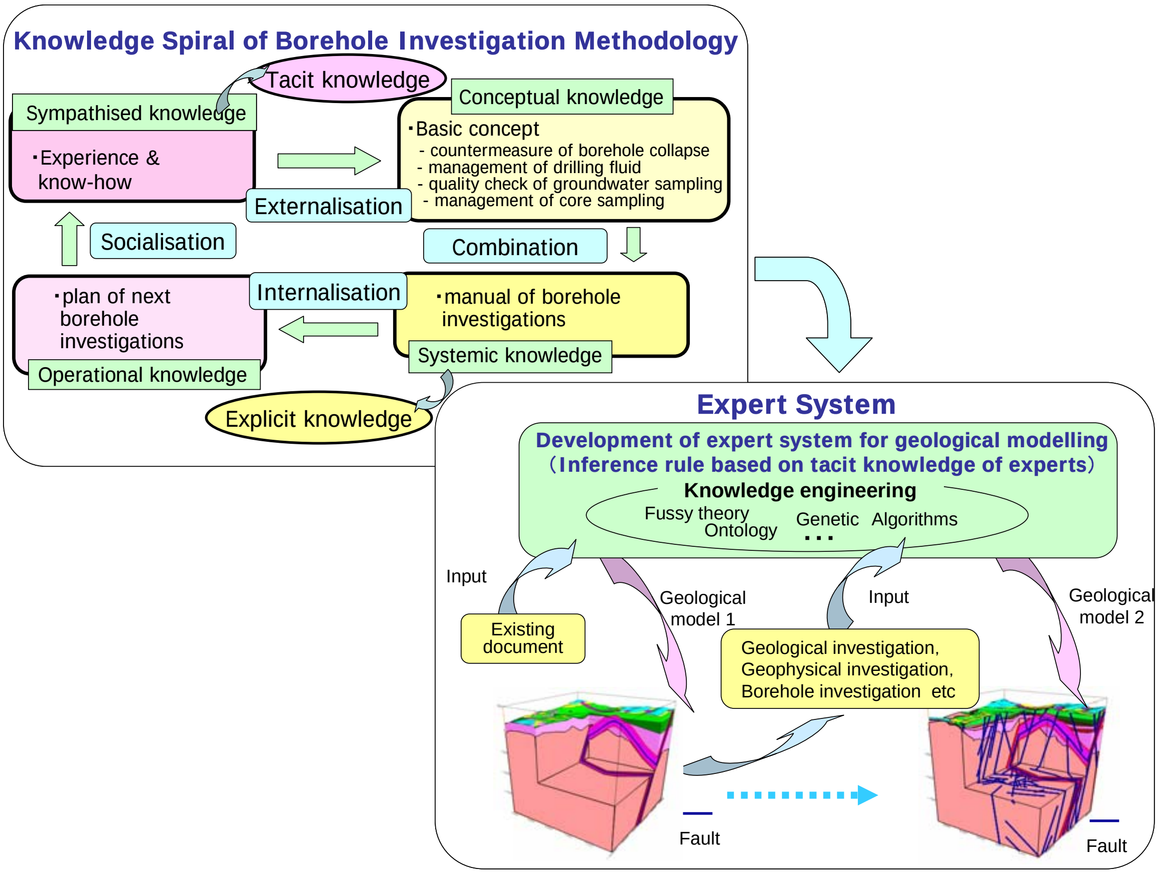


NB: In this figure an element of the safety case, assessment basis, is illustrated as an example for structuring knowledge into knowledge

Another area where innovative approaches are urgently needed for developing and maintaining the knowledge base involves "tacit knowledge" – the experience which exists within a trained workforce. Currently the Japanese programme is in a period of expansion as repository desk studies move towards implementation. This comes at a time when an age bulge is passing through Japanese waste management organisations – the Baby Boomer generation who were associated with the establishment of the nuclear industry in Japan are going into retirement while considerable numbers of young staff are being recruited. This urgent problem will be addressed by both traditional techniques (establishing university courses and training centres where such experience can be passed on) and also more novel methods of preserving experience within expert systems.

An example of such tacit knowledge can be illustrated by considering the considerable amount of expert judgement involved in geological investigation and subsequent conceptual modelling. The "spiral of knowledge creation" [10] terminology has been used to indicate the targets for managing tacit knowledge in the process of borehole investigation, as shown in Figure 3. Conceptual modelling of geological environments is identified as an area where an expert system could be very useful and be developed by establishing rules for inference, based on tacit knowledge of experienced geologists. Knowledge engineering approaches, e.g. establishing ontology, setting up genetic algorithms, etc, can provide a methodological basis for such development. The tacit knowledge effectively results in a "geosynthesis" support tool, which integrates all relevant information on geological environments in a manner suitable for the repository design and safety assessment end-users [11].

Figure 3. **A spiral to distil the tacit knowledge supporting borehole investigation within an expert system for developing geological models**



KMS implementation

The development of this KMS is seen as a long-term process; the KMS will evolve gradually and expand in capabilities as experience is gained in the particular applications noted above. The "electronic assistant" resulting from integration of the software tools mentioned above, in particular, will require development of new technology and/or adaptation of approaches currently used only in unrelated fields. To ensure practicality, specific test cases have been defined, which will allow software developed to be applied at an early stage to subsets of the total safety case. A first workshop to determine the safety-relevant knowledge needs and creation activities focused on two diverse, multidisciplinary areas – assessment of the risk from monogenetic volcanism and evaluation of the consequences of use of low pH cements and concretes. This lead to the identification of some common support tools and approaches which, it is hoped, will allow the benefits of sub-components of the KMS to be directly demonstrated and user feedback to be used to guide future developments. In the future, this will be expanded further with consideration of topics such as definition and use of radionuclide solubility limits and assessment of operational safety.

Initially, at least, the KMS will evolve bilingually in Japanese and English – the former being essential for communication with many stakeholder groups and the latter for communication with the wider international scientific community. As a consequence, a further topic of interest is the development of automatic translation tools, especially for this complex, multidisciplinary area. General developments in this field will be followed, with focused tests of advanced methods (e.g. neutral networks) which could be conditioned by the expanding resources of material available in both languages.

Stepwise implementation is currently planned to proceed over the next 5 years. There certainly will be major challenges being at the front of developments in this area, but these are felt to be balanced by the great gains from early recognition of the essential nature of this challenge and implementation of an efficient response.

Conclusions

The rather ambitious KMS described in this paper is seen as a critical component of the safety cases that will be developed over decades and especially used for licensing towards the middle of the century. Although requiring considerable efforts to develop, the KMS envisaged is considered to be a pragmatic response to a requirement generated by the continuing exponential growth in the quantity of knowledge that supports such cases. The development approach chosen is intended to make tools available to support both knowledge producers and users as soon as possible, with the aim of both demonstrating the practicality of the system and building up support from those who are currently sceptical about its feasibility. In any case, there are certainly generic aspects to this work which might be appropriate for future collaboration between organisations with similar long-term plans for safety case development.

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SAFETY FUNCTIONS AND SAFETY FUNCTION INDICATORS – KEY ELEMENTS IN SKB'S METHODOLOGY FOR ASSESSING LONG-TERM SAFETY OF A KBS-3 REPOSITORY

A. Hedin
SKB, Sweden

Abstract

The application of so called safety function indicators in SKB's safety assessment of a KBS-3 repository for spent nuclear fuel is presented. Isolation and retardation are the two main safety functions of the KBS-3 concept. In order to quantitatively evaluate safety on a sub-system level, these functions need to be differentiated, associated with quantitative measures and, where possible, with quantitative criteria relating to the fulfilment of the safety functions. A safety function is defined as a role through which a repository component contributes to safety. A safety function indicator is a measurable or calculable property of a repository component that allows quantitative evaluation of a safety function. A safety function indicator criterion is a quantitative limit such that if the criterion is fulfilled, the corresponding safety function is upheld. The safety functions and their associated indicators and criteria developed for the KBS-3 repository are primarily related to the isolating potential and to physical states of the canister and the clay buffer surrounding the canister. They are thus not directly related to release rates of radionuclides. The paper also describes how the concepts introduced *i*) aid in focussing the assessment on critical, safety related issues, *ii*) provide a framework for the accounting of safety throughout the different time frames of the assessment and *iii*) provide key information in the selection of scenarios for the safety assessment.

Introduction

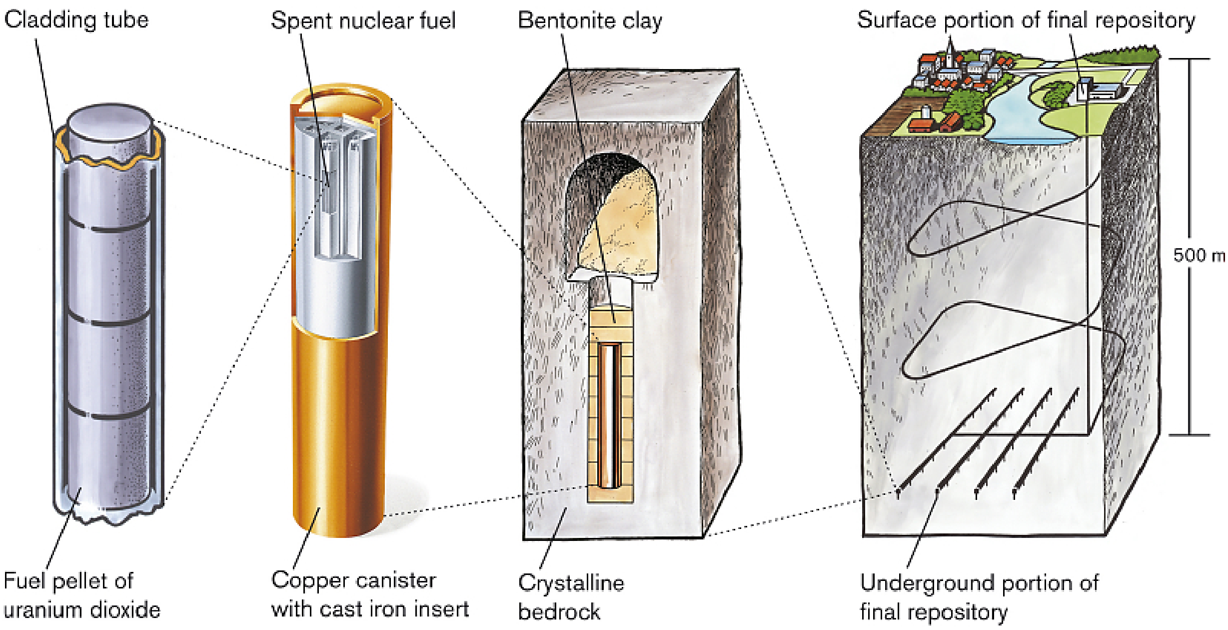
The overall criterion for evaluating long-term safety of a spent nuclear fuel repository in Sweden is a risk criterion issued by the Swedish Radiation Protection Authority, SSI, which states that the annual risk of harmful effects after closure should not exceed one in a million for a representative individual in the group exposed to the greatest risk [1]. The risk calculation is a late step in an elaborate evaluation of repository safety for altering conditions over the one million year assessment period. Both the analysis and the reporting of such an evaluation require intermediate measures that can be used to assess the performance of sub-systems or components of the repository over time.

To meet this need, so called safety function indicators have been introduced in SKB's safety assessments for spent nuclear fuel repositories. A safety function indicator is a measurable or calculable property of a repository component that is related to a safety function of that component. In many cases, it is also possible to define a function indicator criterion, such that the function is acceptably upheld as long as the criterion is fulfilled.

The first full application of safety function indicators is found in SKB's safety assessment SR-Can, published in November 2006 [2]. The assessment is an important step towards the SR-Site assessment, to be delivered in 2009, which will support a licence application for a Swedish final repository for spent nuclear fuel.

The assessment relates to the KBS-3 disposal concept in which copper canisters with a cast iron insert containing spent nuclear fuel are surrounded by bentonite clay and deposited at approximately 500 m depth in saturated, granitic rock, Figure 1. Preliminary data from SKB's on-going investigation of two candidate sites for a final repository for spent nuclear fuel are used in the SR-Can assessment.

Figure 1. **The KBS-3 concept for final storage of spent nuclear fuel**



Definition of function indicators and function indicator criteria

The primary safety function of the KBS-3 concept is to completely isolate the spent nuclear fuel within copper canisters over the entire assessment period. Should a canister be damaged, the secondary safety function is to retard any releases from the canisters. It is noted that the isolation function is more prominent in the KBS-3 concept than in several other repository concepts for spent nuclear fuel or high level waste, e.g. [3,4]. This is also reflected in the methodology and structure of the safety assessment, which focuses to a comparatively large extent on the isolating capacity of the repository.

Safety functions

Understanding and evaluating repository safety in a detailed and quantitative manner requires a more elaborate description of how the main safety functions isolation and retardation are upheld by the components of the repository. Based on the understanding of the properties of the components and the long-term evolution of the system, a number of subordinate safety functions to isolation and retardation can be identified.

In this context, a safety function is defined qualitatively as a role through which a repository component contributes to safety. For example, copper corrosion could in the long term jeopardise the isolation function of the canisters. Certain species of microbes in the buffer could transform sulphate to sulphide, a copper corroding agent. A safety function related to the buffer and subordinate to isolation would therefore be the ability of the buffer to prevent microbes from surviving in it.

Safety function indicators

In order to quantitatively evaluate safety, it is desirable to relate or express the safety functions to measurable or calculable quantities, often in the form of barrier conditions.

In the case of the buffer function of eliminating microbes, current understanding indicates that a high buffer swelling pressure prevents the survival of microbes. The buffer swelling pressure is thus a suitable quantity to use in order to evaluate the extent to which this function is fulfilled. The buffer swelling pressure is said to be a safety function indicator for the mentioned buffer function.

A safety function indicator is thus a measurable or calculable quantity through which a safety function can be quantitatively evaluated.

Safety function indicator criteria

In order to determine whether a safety function is upheld or not, it is desirable to have quantitative criteria against which the safety function indicators can be evaluated.

The situation is, however, different from safety evaluations of many other technical/industrial systems in an important sense: The performance of the repository system or parts thereof do not, in general, change in discrete steps, as opposed to e.g. the case of a pump or a power system that could be characterised as either functioning or not. The repository system will evolve continuously and in many respects there will be no sharp distinction between acceptable performance and a failed system on a sub-system or regarding detailed barrier features.

There are thus many safety function indicators for which no limit for acceptable performance can be given. The groundwater concentrations of canister corroding agents or agents detrimental to the buffer are examples of this kind of factor related to isolation. Usually, they enter in more complex analyses where a number of parameters together determine, e.g., the corrosion rate of the canister. Most of the factors determining retardation are also of this nature.

Nevertheless, as will be demonstrated below, there are some crucial barrier properties on which quantitative limits can be put. Regarding isolation, an obvious condition is the requirement that the copper canister should nowhere have a penetrating defect, i.e. there should, over the entire surface of the canister, be a non-zero copper thickness. In addition to this direct measure of isolation performance, a number of quantitative supplementary criteria can also be defined. These relate, for example, to the peak temperature in the buffer and to requirements on buffer density and buffer swelling pressure giving favourable buffer properties for maintaining isolation. Most of them determine whether certain potentially detrimental processes can be excluded from the assessment. Relating to the above example of microbes in the buffer, preliminary data suggest that the buffer should have a swelling pressure higher than 2 MPa for microbial survival in the buffer to be excluded. The requirement that the buffer swelling pressure should exceed 2 MPa is thus a safety function indicator criterion in this case (provided that the preliminary data are confirmed).

A safety function indicator criterion is thus a quantitative limit such that if the function indicator to which it relates fulfils the criterion, the corresponding safety function is upheld.

Relation between global safety and individual safety functions

It is emphasised that the breaching of a safety function indicator criterion does not mean that the repository is unsafe, but rather that more elaborate analyses and data are needed in order to evaluate safety.

The criteria are an aid in determining whether safety is maintained. If the criteria are fulfilled, the safety evaluation is facilitated, but fulfilment of criteria alone is not a guarantee that the overall risk criterion is fulfilled. On the other hand, compliance with the risk criterion could well be compatible with a violation of one or several of the safety function indicator criteria. A violation would be an implication of caution; further analyses could be required in order to determine the consequences on a sub-system level or a system level.

An example is the criterion that the groundwater concentration of divalent cations should exceed 1 mM in order for buffer erosion to be excluded. If this criterion is breached, buffer erosion must be quantitatively evaluated and its consequence in terms of reduced buffer density needs to be propagated to assessments of buffer swelling pressure and hydraulic conductivity. Alterations of the latter factors could, in turn, influence e.g. canister corrosion. A chain of assessments is thus initiated by the breaching of the first safety function, but the final outcome of a possibly increased corrosion rate is not necessarily an unacceptable impact on isolation.

Miscellaneous issues related to safety functions

The following issues related to the safety functions used in SR-Can are further discussed in the SR-Can Main report [2]:

- The different function indicator criteria are determined with varying margins to acceptable performance.
- Safety functions are related to, but not the same as, design criteria. Whereas the latter relate to the initial state of the repository and primarily to its engineered components, the former should be fulfilled throughout the assessment period and relate, in addition to the engineered components, to the natural system.
- The set of safety functions used in SR-Can are currently being derived through a systematic approach applied to the documented experiences accumulated over decades of research related to the long-term safety of the KBS-3 repository.
- The safety functions are related. All safety functions of the buffer either support a safety function of the canister, or contribute to retardation in the buffer. Similarly, all safety functions of the host rock either support a safety function of the canister directly or indirectly via a buffer safety function, or contribute to retardation in the rock.

Function indicators for the KBS-3 repository

A preliminary set of function indicators for a KBS-3 repository is presented in Figure 2. This set is mainly related to the isolation function.

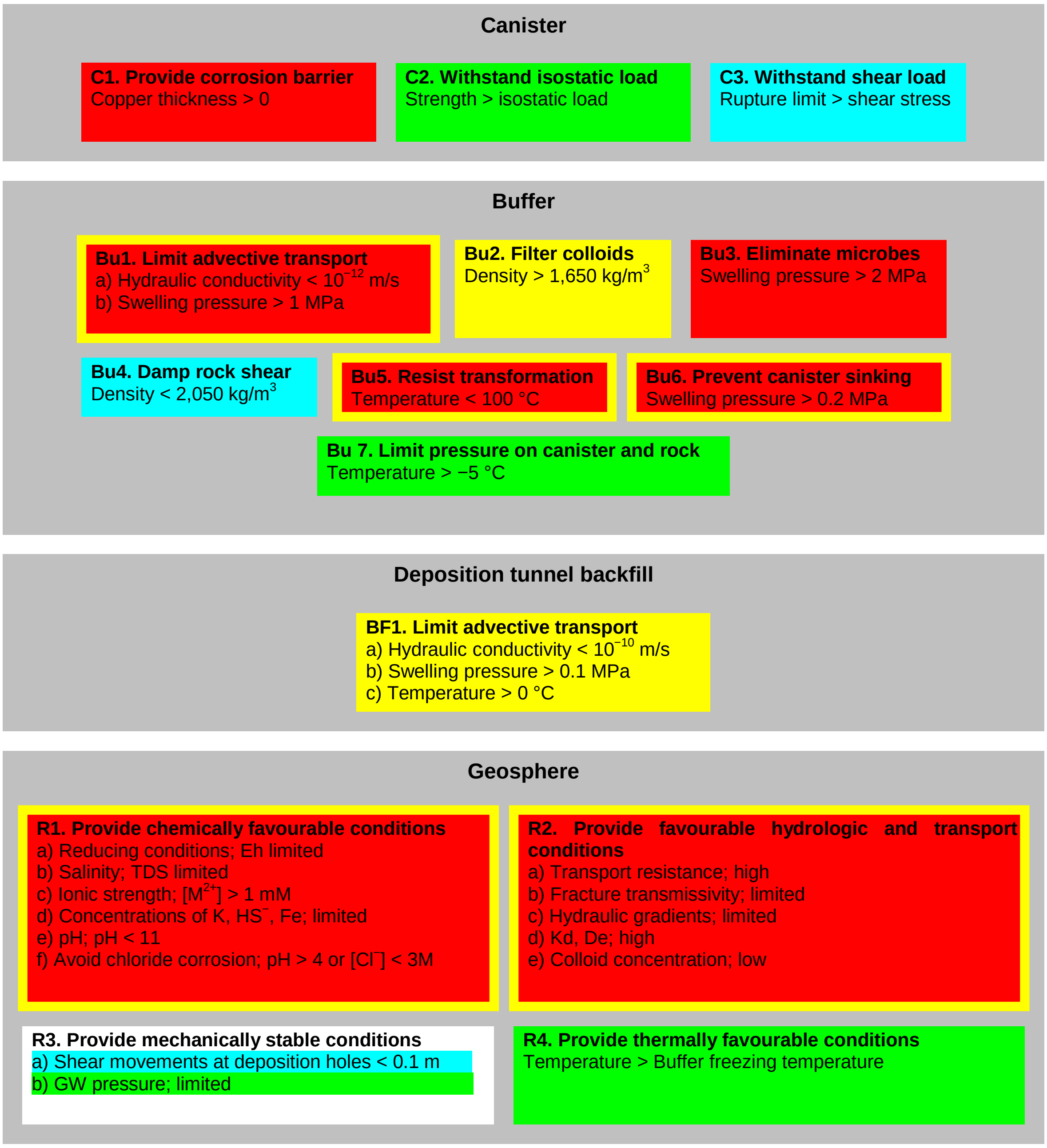
For the canister, the function indicator C1 in Figure 2 is related to the main function of the copper shell, namely to provide a corrosion barrier. The function indicators C2 and C3 are related to the mechanical functions of the canister insert, namely to withstand isostatic loads and shear loads, respectively.

The main role of the bentonite buffer is to limit advective transport, i.e. to ensure that diffusion is the dominating mechanism for both inward transport of canister corroding agents in the groundwater and potential outward transport of radionuclides. This is achieved if the buffer has a sufficient hydraulic conductivity, Bu1a, and a sufficient swelling pressure, Bu1b. The latter requirement is imposed in order to ensure buffer tightness through self-healing and reswelling. The buffer should also hinder the outward migration of fuel colloids, requiring a certain density, Bu2. Furthermore, the buffer should prevent microbes from surviving in it, requiring a certain swelling pressure, Bu3 (value to be finally established). Note that this requirement is imposed in order to ensure that microbes initially present in the buffer do not survive, whereas the pore size also at very low densities will prevent additional microbes from entering the buffer from the host rock. The buffer's ability to damp shear movements in fractures intersecting the deposition hole is related to its mechanical properties, which are in turn largely controlled by its density, Bu4. A temperature limit is imposed to be able to exclude mineralogical transformations of the buffer, Bu5. Another role of the buffer is to prevent the canister from sinking to the bottom of the deposition hole, which is ensured through a certain swelling pressure, Bu6. Finally, the buffer should not freeze, since this could imply considerable mechanical forces on the canister and the surrounding rock. This is ensured if the temperature exceeds the freezing temperature of the buffer, −5°C.

The main function of the deposition tunnel backfill is to limit advective transport, which is achieved through requirements on the backfill hydraulic conductivity and swelling pressure, functions BF1a and BF1b, respectively. Also, the backfill should prevent upward expansion of the buffer. This has, however, not led to a formulation of a function indicator since a violation of such a function would lead to violations of several of the buffer functions and thus be captured through these. It is, though, an important design requirement when selecting backfill materials.

For the geosphere, the safety functions can be grouped into four categories related to favourable *i*) chemical (R1), *ii*) hydrologic and transport (R2), *iii*) mechanical (R3), and *iv*) thermal (R4) conditions. Most of these can be further differentiated into more detailed requirements, see Figure 2. However, only in a few cases has it been possible to formulate quantitative safety function indicator criteria. Important examples relate to the concentration of divalent cations in the groundwater that needs to exceed a certain value in order to prevent colloid formation and associated erosion of the buffer, and to rapid rock shear movements at deposition holes that need to be limited in order to avoid mechanical failure of the canister.

Figure 2. **Safety functions (bold), safety function indicators and safety function indicator criteria. When quantitative criteria cannot be given, terms like “high”, “low” and “limited” are used to indicate favourable values of the safety function indicators. The colour coding shows how the functions contribute to the canister safety functions C1 (red), C2 (green), C3 (blue) or to retardation (yellow). Many functions contribute to both C1 and retardation (red box with yellow board).**



Application of safety functions in the safety assessment

The safety functions with their indicators and associated criteria assist in the analysis essentially in three ways.

First, they provide an early identification of critical issues to be studied in the safety assessment.

Second, in the analysis of a comprehensive **main scenario** describing a **plausible evolution** of the repository system, the function indicators provide a structure for evaluating safety. The repository evolution is analysed in a number of timeframes and for each timeframe, safety is systematically evaluated through an account of the status of the function indicators during and at the end of that timeframe.

Third, the functions and function indicators are used in the derivation of additional scenarios for the evaluation of uncertainties not taken into account in the main scenario.

The two latter roles of the safety functions and the safety function indicators are further described in the two following sub-sections.

Structure for evaluating safety in the main scenario

A reference evolution of the repository over the entire one million year assessment period is studied to gain an understanding of the overall evolution of the system as a basis for scenario selection and scenario analyses. The aim is to describe a reasonable evolution of the repository system over time. The reference evolution forms the basis for the comprehensive main scenario in the assessment. In the reference evolution/main scenario, the external conditions during the first 120 000 year glacial cycle are assumed to be similar to those experienced during the last cycle, the Weichselian. Thereafter, seven repetitions of that cycle are assumed to cover the entire one million year assessment period. Initially, immediately after deposition, the repository is assumed to be in a reference initial state, defined in accordance with specifications on engineered barriers etc, taking into account the quality of the procedures for manufacturing of engineered barriers, for rock excavation, etc.

The analysis of the reference evolution is carried out in four time frames:

- The excavation/operational period.
- The initial period of temperate domain from the reference glacial cycle.
- The remaining part of the initial glacial cycle.
- Subsequent glacial cycles up to one million years after closure.

The reporting of the analysis in each time frame concludes with a discussion of the expected status of the safety function indicators during and at the end of the time frame. For example, the results of the comprehensive analyses of thermal, hydraulic, mechanical and chemical phenomena during the third time frame, the remaining part of the initial glacial cycle after the first temperate period, imply the following:

- Large earthquakes, of magnitude 6 or larger, in the vicinity of the repository are highly unlikely but cannot be completely ruled out. Results of probabilistic calculations imply that the mean number of canister failures during the initial glacial cycle due to such events is of the order of 10^{-2}. This relates to safety functions C3 and R3a in Figure 2.

- Dilute groundwaters may occur for extended periods of time when glacial conditions prevail. This may lead to loss of buffer mass in some deposition holes, to the extent that advective conditions are created. This leads to enhanced canister corrosion, but no canisters are assessed to fail during the initial glacial cycle. This relates mainly to the safety functions C1, Bu1 and R1c in Figure 2.

Other aspects of the evolution during this time frame are assessed not to threaten any of the safety functions of the repository.

Derivation of additional scenarios

The further assessment of repository safety is broken down into a number of additional scenarios. A set of such scenarios are defined in order to cover uncertainties not addressed in the main scenario, e.g. more extreme climate conditions than those obtained from the reconstruction of the Weichselian glacial cycle in the main scenario.

The safety functions are used to obtain a comprehensive set of additional scenarios, focussing on issues of relevance to repository safety. When defining a scenario, a violation of a safety function is *postulated* and all conceivable routes to such a violation are then scrutinised. The aim is to answer the question: Is there any reasonable way in which this scenario could occur? If this is found to be the case, the consequences of the scenario in question are included in a risk summation for the repository. If not, the scenario is considered as “residual”, and consequences may be analysed for illustrative purposes. For each selected scenario, uncertainties related to initial state factors, processes and external conditions that are not covered in the main scenario are considered. The analysis of the main scenario forms an important starting point for the analysis of each of these additional scenarios.

For example, a scenario with canister failures due to isostatic collapse, relating to function indicator C2 in Figure 2, is considered. This failure mode is ruled out as a risk contributor in the main scenario where a plausible evolution is analysed. In the canister failure scenario, all possible routes to this failed state are critically evaluated, including assessments of the most unfavourable external conditions, in this case pressure from a glacier overburden of maximum thicknesses, and initial conditions, in this case extreme canister manufacturing flaws. The aim is to determine whether the scenario should be assigned a finite probability or whether it could be ruled out as a risk contributor and only analysed as a pure “what if” scenario.

A set of scenarios is selected such that all function indicators are covered. Several of the safety functions are though of an overlapping or complementary nature. Therefore, in some cases, several of the safety functions are lumped into the same scenario. The scenarios related to function indicators in SR-Can are:

- Advective conditions in the buffer (Bu1a and Bu1b).
- Buffer freezing (Bu7).
- Buffer transformation (Bu5).
- Canister failure due to isostatic load (function indicator C2 in Figure 2).
- Canister failure due to shear load (C3).
- Canister failure due to corrosion (C1).

The scenarios related to the buffer safety functions are analysed first and then combined with each of the scenarios related to canister failures.

Thus, in the comprehensive analysis of the reference evolution/main scenario, a number of coupled processes/phenomena are identified and their impact on the safety functions for reference conditions are analysed. When analysing the additional scenarios, all these coupled processes/ phenomena are revisited, but now for the more extreme conditions considered in the additional scenarios. The analysis of the main scenario thus has an important role also in providing an identification of issues to address in the additional scenarios. Furthermore, by focusing each additional scenario on one particular safety function, the number of phenomena to consider is reduced, thereby limiting the complexity of the analysis.

Summary and conclusions

The following definitions have been introduced:

- A safety function is a role through which a repository component contributes to safety.
- A safety function indicator is a measurable or calculable property of a repository component through which a safety function can be quantitatively evaluated.
- A safety function indicator criterion is a quantitative limit such that if the function indicator to which it relates fulfils the criterion, the corresponding safety function is upheld.

Safety functions are an aid in evaluation safety, but the fulfilment of all safety function indicator criteria is neither necessary nor sufficient to argue safety. The different function indicator criteria are furthermore determined with varying margins to acceptable performance.

Safety functions are related to, but not the same as, design criteria. Whereas the latter relate to the initial state of the repository and primarily to its engineered components, the former should be fulfilled throughout the assessment period and relate, in addition to the engineered components, to the natural system.

The function indicators have proven useful in the safety assessment SR-Can i) by aiding in the focusing, at an early stage, on critical issues to be studied in the safety assessment, ii) by providing a structure for evaluating safety in a comprehensive main scenario, and iii) by providing key information in the selection of additional scenarios in the assessment.

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THE ROLE OF STRUCTURAL RELIABILITY OF GEOTECHNICAL BARRIERS OF AN HLW/SF REPOSITORY IN SALT ROCK WITHIN THE SAFETY CASE

N. Müller-Hoeppe, M. Jobmann, M. Polster, H. Schmidt
DBE TECHNOLOGY GmbH, Germany

Abstract

Regarding a final repository in salt rock geotechnical barriers are placed parallel to the impermeable geological rock salt barrier to seal the access tunnels and shafts in order to isolate the disposal areas from the biosphere. Geotechnical barriers are shaft and drift seals, and in some cases borehole seals. Their function is to efficiently seal the man-made pathways directly after repository closure. As the proof of structural reliability of geotechnical barriers is fundamental to assess the long-term safety of an HLW/SF final repository in salt rock, the present status of the proof of structural reliability of geotechnical barriers was reviewed. The result of the review process is given below.

Introduction

Preliminary Total Performance Assessments for HLW repositories in salt rock were performed within the framework of German R&D projects at the end of the 1980s and in the first half of the 1990s. In the meantime, several remarkable developments improved the basis for developing an advanced safety case:

- Surface and particularly underground exploration results of the Gorleben salt dome are available and provide an improved understanding of the geological system and corresponding processes.
- In the course of developing the closure concepts for the Morsleben Repository and Asse Mine, major progress was made regarding the design and performance proof of engineered barriers, in particular shaft and drift seals.
- Advanced tools have been developed and the knowledge base for assessing the consequences of radionuclide release in case of disturbed repository evolution has significantly improved.
- Contrary to the existing German safety criteria, the ongoing revision of these criteria intends to classify possible repository evolution incidents into scenarios with high and low probabilities.

In view of these changes, DBE TECHNOLOGY GmbH, BGR and GRS are developing and testing an advanced safety concept within a joint R&D project "Überprüfung und Bewertung des bereits verfügbaren Instrumentariums für eine sicherheitliche Bewertung von Endlagern für HLW" (ISIBEL) in order to identify the major needs for further R&D work.

Regarding a final repository in salt rock geotechnical barriers are placed parallel to the impermeable geological rock salt barrier to seal the access tunnels and shafts in order to isolate the disposal areas from the biosphere. Geotechnical barriers are shaft and drift seals, and in some cases borehole seals. Their function is to efficiently seal the man-made pathways directly after repository closure. In this context, crushed salt backfill is not considered as a geotechnical barrier as due to its high initial porosity, its sealing capability is mainly of long-term importance. However, it plays a fundamental role in the design of geotechnical barriers.

As the proof of structural reliability of geotechnical barriers is fundamental to assess the long-term safety of an HLW/SF final repository in salt rock, the present status of the proof of structural reliability of geotechnical barriers was reviewed within the scope of the ongoing ISIBEL project. Experience from rock salt mining was taken into account as well as new results from closing the German LILW repository "Endlager für radioaktive Abfälle Morsleben" (ERAM) and the Asse mine. The proof of structural reliability of conventional geotechnical barriers served as a basis for comparison.

The efficiency of conventional barriers throughout their design life has to be shown using quantitative design criteria. For this kind of conventional barriers, the proof of function (e.g. tightness) is required to be state of the art. Presently, state-of-the-art technology consists of the following proofs:

- Proof of tightness or proof of adequate flow resistance.
- Proof of mechanical resistivity (e.g. mechanical stability, limited crack evolution).
- Proof of durability.
- Proof of producibility.

State-of-the-art technology includes the proof of practical applicability with respect to the required functions. Determining the state-of-the-art technology, comparable structures and construction procedures, which have been tested successful in practice, should be taken into account. Concerning conventional geotechnical barriers, state-of-the-art technology is defined for instance by the system of Structural Eurocodes and the related national standards. The application of these technical guidelines is specifically permitted for conventional barriers of near-surface and surface disposal sites. If these regulations are applied the efficiency of a conventional geotechnical barrier is shown with an adequate degree of reliability that is quantitatively defined.

When assessing the safety of a final repository for radioactive waste quality assurance is often put on a level with structural reliability. However, quality assurance is only one aspect of structural reliability.

The Gorleben working model – an HAW/SNF final repository in salt rock

In the Gorleben working model [1], the geological rock salt barrier of the well-designed repository is assumed to be intact. Safety criteria and methods to assess and prove the functionality (integrity) of a rock salt barrier are available. Moreover, they are applied in the context of the closure of the Morsleben repository (ERAM) [2].

Intact rock salt has several inherent safety functions, i.e.:

- Rock salt areas suitable for the disposal of radioactive waste are dry – mobile water content is negligible
- Intact rock salt shows a negligible pore space – no reservoir for brines or other liquids is available
- The pores of intact rock salt are not interconnected, forming an effective flow barrier

Thus, the crushed salt backfill as provided in the Gorleben working model is also taken into account because of its sealing and healing capability. In the case of the Gorleben working model, the safety design [3] focuses on restoring at least one of the safety functions of its formerly intact rock salt condition by means of engineered barriers. Then, radionuclide transport is effectively prevented because either the transport medium is missing, or the transport path, the potential to dissolve radionuclides or the transport mechanisms (gas production, convergence) are being suppressed.

The time dependent behaviour of crushed salt backfill under repository conditions is a key parameter in deriving requirements for geotechnical barriers. The engineered geotechnical barriers acting in parallel to the geological barrier are to be designed in such a way that at least one of the natural rock salt safety functions is restored even if the repository evolution is disturbed.

In order to develop design requirements for geotechnical barriers indicative calculations were performed to assess the brine in- and outflow using a conceptual model of a simplified repository layout and of the multi-barrier system consisting of the shaft seals, drift seals and backfill without detailed specification of the disposal areas [4]. Three scenarios were investigated: Two intact geotechnical barriers (undisturbed evolution), shaft seal failure (disturbed evolution), and drift seal failure (disturbed evolution) [3].

The calculation results can be summarised as follows:

- When using parameter values of the reference case and investigating the three scenarios mentioned above both geotechnical barriers were permeated by brine, however, in all cases the amount of brine was less than 1 m^3, neither filling the residual pore space of the access drift to the disposal areas nor the disposal areas. Even in case of unfavourable variations the amount of brine inflow remained below 2 m^3. Thus, no radionuclide release occurs. However, the calculation results are based on an assumed geotechnical barrier working life of a minimum of 100 000 years.
- Next, the time period until additionally established borehole plugs consisting simply of crushed salt have reduced their porosity to the lower limit of 0.001 was calculated for the special case of borehole disposal. Depending on the type of waste emplaced in the borehole, the lower porosity limit is reached within a time period of 50-600 years. As boreholes are dead-end structures enclosed by the intact geological rock salt barrier, no brine will flow through. Thus, the amount of brine for dissolving radionuclides is crucially restricted by the limited pore space available.

Fundamental requirements for geotechnical barriers

In the context of long-term safety assessment, only one parameter of the geotechnical barrier is of interest, its flow resistance over time. Mostly for practical reasons, the flow resistance is described as a function of the geotechnical barrier's length, its cross-sectional area including the excavation damaged zone (EDZ), and its permeability. The parameters used for indicative calculations in case of the Gorleben working model (reference case) are given below [3]:

Length: 50 m	Permeability: 10^{-18} m^2
Cross section shafts (accumulated): 44 m^2	Cross section drifts (accumulated): 110 m^2

In geotechnical barrier design, the design life is an important quantity. If the required design life does not exceed 50 - 100 years, the geotechnical barriers may be designed using state-of-the-art technology according to the European Standard [5] and related national guidelines [6,7,8].

In the following this case will be regarded first and subsequently, a prolongation of the design life will be considered.

State of the art regarding geotechnical barriers

First, state-of-the-art engineering to avoid brine intrusion is considered. At an early stage, brine intrusion is prevented by geotechnical barriers such as shaft and drift seals.

Focusing on shaft and drift seals and taking into account European Standards in civil engineering [5], a technical structure of safety relevance is designed to have an upper failure limit of 10^{-6}/a, where the working life of the technical structure is assumed to be 50 - 100 years, i.e. a failure probability of $\leq 10^{-4}$/a holds for the working life.

According to the European Standard [5], a structure is to be designed and executed in such a way that during its intended life it will – with an adequate degree of reliability and in an economical way:

- Sustain all actions and influences likely to occur during execution and use, and remain suitable for the use for which it was designed.
- Not be damaged by accidential events respectively impacts or the consequences of human errors to an extent inappropriate to the original cause.

In [5], informative annexes are available describing how to proceed to guarantee structural reliability. Three annexes are of special interest because they are applicable to geotechnical barriers:

(1) Management of structural reliability for construction works.
(2) Basis for partial factor design and reliability analysis.
(3) Design assisted by testing.

(1) Focuses mainly on the aspects of reliability differentiation, design supervision differentiation, and inspection during execution. These aspects are not treated in this paper. In the partial factor method (2) the basic variables (actions, resistances, and geometrical properties) are given design values by using partial factors and combination factors and a verification is made to ensure that no relevant limit state has been exceeded. In principle, numerical values for partial factors and combination factors can be determined on the basis of calibration to the long-term experience of the building tradition or on the basis of statistical evaluation of experimental data and field observation. In practice, the partial factor design method constitutes a method to prove structural reliability mainly by calculation. It was applied when designing the ERAM drift seals. An equivalent method to partial factor design is the design assisted by testing method (3). In some cases testing may be carried out, e.g. if adequate calculation models are not available or to confirm assumptions made in the design. Design assisted by testing is based on a combination of tests and calculations whereas testing plays a fundamental role. This procedure was applied when designing the Salzdetfurth Shaft II seal.

ERAM drift seals

In the case of the ERAM seals [2,9,10,11] the drift seals' lengths as well as their cross-sectional area is determined by local conditions. Thus, the average permeability is the design criterion directly related to long-term safety. An average permeability limit of 10^{-18} m^2 is required. There are two reasons why it might be exceeded:

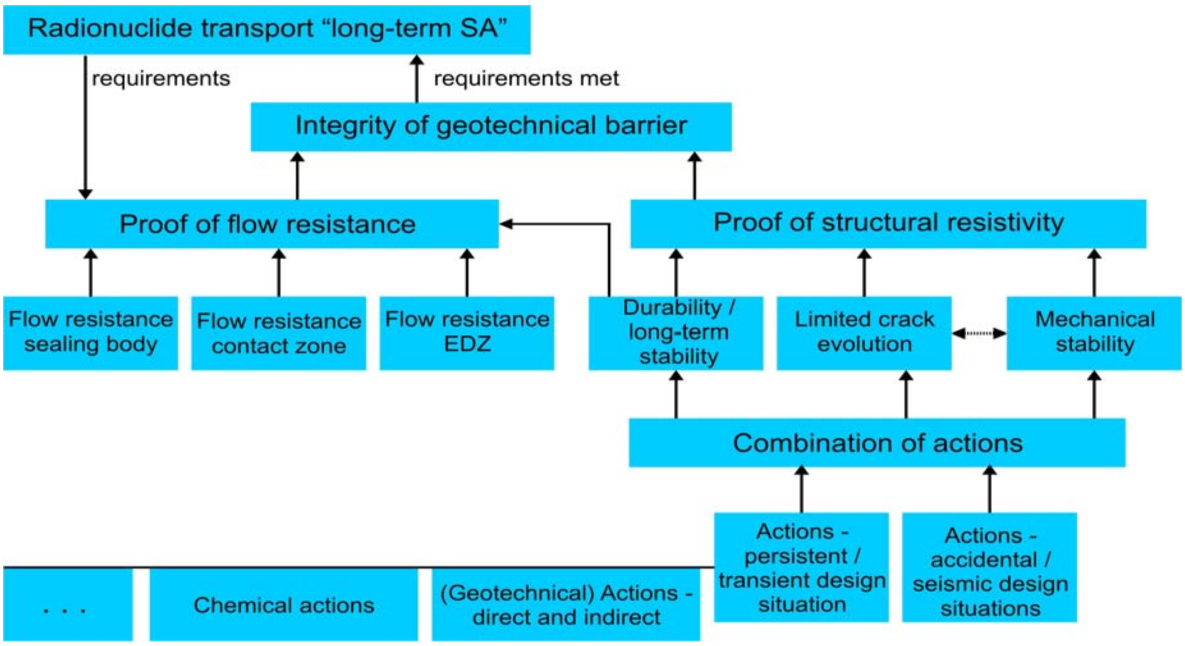
- mechanical failure and crack evolution.
- insufficient flow resistance of the sealing body, the contact zone, or the EDZ.

Mechanical stability and integrity (limited crack evolution) have to be proven taking into account several design situations, i.e. during the construction phase as well as under dry and wet repository conditions in the long term. The interaction between sealing body, contact zone, and EDZ must be considered because they form a firm unit. Flow resistance, however, is of importance only under wet repository conditions. If the flow resistance proves to be sufficient the sealing body, the contact zone, and the EDZ may be treated separately.

The proof of durability or long-term stability is also coupled with the proof of adequate flow resistance, as low alteration rates are a direct consequence of high flow resistance. The coupling of proofs related to different limit states are given in Figure 1.

In case of the ERAM drift seals the individual proofs related to different limit states and design situations are described in [11]. As a result it can be stated that the method of partial factor design was applied in order to guarantee an adequate level of reliability. Most of the proofs were successfully performed. Still under discussion is the proof of limited crack evolution during the construction phase (transient design situation) and the hydraulic resistance of the EDZ in case of high brine pressure. Regarding the structural reliability of the whole structure, the EDZ has the highest uncertainty factor at present [11].

Figure 1. **Scheme of proofs to demonstrate geotechnical barrier integrity**



Salzdetfurth Shaft II seal

In Germany, the mining authority sometimes demands that the operator of a salt mine keeps it in dry condition after closure, i.e. the operator is to prevent water respectively brine intrusions into the mine. The R&D project Salzdetfurth Shaft II seal was performed to prove the long-term dry closure of a conventional salt mine. This proof was required to exclude any harm to the Salzdetfurth town. Regarding the Gorleben working model, the main task of the shaft seals is to prevent brine intrusion into the repository mine in case of undisturbed repository evolution. Thus, the results of the Salzdetfurth Shaft II seal project are essential when determining state of the art in designing shaft seals. In case of the Salzdetfurth Shaft II seal the alternative approach to prove structural reliability was used, i.e. the design assisted by testing method was used. The large scale *in situ* test was designed to determine the ultimate resistance of the whole structure for given load conditions.

Prior to carrying out the tests the European Standard [5] requires that a test plan is established, which is to be cleared with the testing organisation. This plan is to contain the objectives of the test and all specifications necessary for the selection or production of the test specimens, the execution of the tests, and the test evaluation. The test plan should cover:

- Objectives and scope.
- Prediction of test results.
- Specification of test specimens and sampling.
- Loading specifications.
- Testing arrangement.
- Measurements.
- Evaluation and reporting of tests.

For the Salzdetfurth Shaft II seal, the objectives and scope were defined, i.e. mechanical resistivity, adequate flow resistance, maintenance-free structure, and state-of-the-art producibility.

After testing the load bearing capactiy and subsidence resistance of the lower gravel abutment individually, the final test was conducted in an experimental shaft of 2.50 m diameter.

Quantitatively, adequate flow resistance is defined by an average hydraulic conductivity of $k_f \leq 5 \cdot 10^{-10}$ m/s in case of NaCl-brine and a short-time bentonite swelling pressure of 1 MPa. As a main action, brine pressure is applied which is linearly increased to 4 MPa within 14 days and subsequently to 7 MPa within a further 7 days in order to approximate an exponentially increasing brine intrusion.

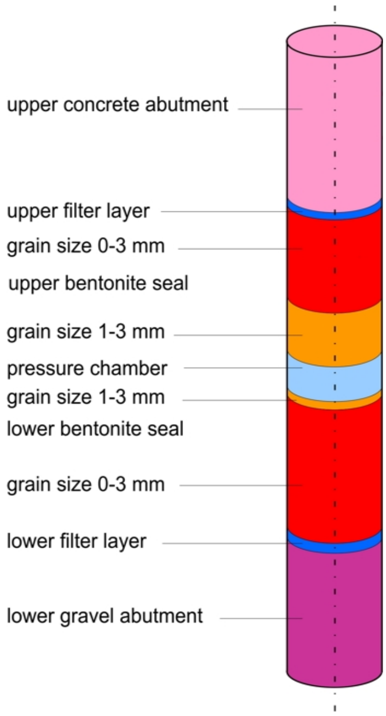
The *in situ* experiment was performed inside an experimental shaft, the test arrangement is schematically shown in Fig. 2. The predicted test results are described in [12] as well as small- and medium-scale laboratory tests which were performed prior to the *in situ* test. The objective of the test set-up was to measure the bentonite swelling pressure *in situ*, the deformations of bentonite under load, the saturation and the brine front inside the bentonite seal and at the bentonite/rock salt interface.

In the following, the experimental results are summarised. The description of the test results is restricted to the lower bentonite seal because it is representative for the design situation of a dry repository as regarded for the Gorleben working model in the ongoing ISIBEL project.

The average hydraulic conductivity was measured in the range of $k_f \leq 5.8 \cdot 10^{-11}$ m/s (4 MPa brine pressure) and $k_f \leq 4.4 \cdot 10^{-11}$ m/s (7 MPa brine pressure). The swelling pressure determined in situ showed a range of 1.0 MPa – 1.2 MPa.

When the in situ measurements had been concluded the lower bentonite seal was dismantled. The investigation of the position of the brine front and the saturation state confirmed the results measured before and showed good agreement with results predicted by calculation. Thus, the calculation model was validated additionally. Finally, it can be concluded that the experimental shaft experiment was successfully performed a more detailed report is given in [12].

Figure 2. **Schematic test arrangement of the experimental shaft *in situ* experiment**



At the level of interpretation of test results, one has to keep in mind that only one test was performed so that no classical statistical intepretation is possible [5]. The result of the test evaluation should be considered to be valid for the specifications and load characteristics considered in the tests only. If the results are to be extrapolated to cover other design parameters and loads, additional information from previous tests or from theoretical sources should be used. This work is not finished yet and further work is needed to evaluate structural reliability. First results are described in [13] focusing on a restricted number of limit states. The failure probability determined [13] showed a range of 10^{-4} to $6{\cdot}10^{-3}$ per working life. The upper bound, however, is based of insufficient data.

Although the data base has to be improved, the results achieved so far are very promising. For this reason, a comparable seal is planned to seal the shafts of the Asse experimental mine.

Prolongation of design life

In the case of the ERAM drift seals, the proof of structural reliability is determined by the early stage design situation except when proving durability or long-term stability. As a design life of 5 000-30 000 years is required, the ERAM drift seals' prolongation of design life was achieved by detailing the proof of durability. Alteration of salt concrete was taken into account and the consequences of alteration were checked based on experimental results which were extrapolated into the future by numerical modelling [14,15]. As the alteration did not significantly change the properties of the structure within the required design life, the influence of alteration on other proofs is negligible.

In case of the Salzdetfurth shaft seals, durability/long-term stability was shown using saliferous clay as a natural analogue to bentontite [16].

Proof of producibility

For the ERAM seals the proof of producibility is still pending. Producibility will be proven in the framework of a pilot study. However, the Asse seal [11,17] constitutes an example of producibility, although some lessons had to be learnt regarding the properties of the contact zone at the roof.

For the Salzdetfuth Shaft II seal producibility was successfully proven by the experimental shaft project [12]. Meanwhile, the bentonite seal of Salzdetfurth Shaft II has already been built in the context of the dry closure of the Salzdetfurth mine [18].

Summary

With regard to geotechnical barriers the present state of dry closure of an HLW/SF repository may be summarised as follows:

The so-called dry closure against brine intrusion is state of the art when dry-closing conventional salt mines. The required proof is state-of-the-art, too.

Presently, dry closure of conventional salt mines is guaranteed by adequate shaft seal design. A prototype seal was successfully tested in the Salzdetfurth mine. The data base has to be improved to ensure structural reliability.

As the mining regulations applied in Germany guarantee a tight geological barrier in modern mines there was no need to build tight drift seals against an actual brine intrusion for several decades. Thus, an adequate drift seal design is discussed mainly under theoretical aspects supported by *in situ* investigations. However, a method to prove structural reliability has been developed. Most of the individual proofs have been performed successfully, some of them are in progress showing promising results. The example of the Asse seal demonstrates producibility in general.

Finally, it may be concluded that in order to ensure the dry closure of an HLW/SF repository in accordance with safety design aspects the construction of two independent geotechnical barriers – shaft seals and drift seals – showing different designs and adequate structural reliability (failure $\leq 10^{-4}$ per working life) according to [5] is promising.

The Morsleben repository and the Asse mine are both mines that had been used for mineral extraction before radioactive waste was emplaced. In both cases safe enclosure is not considered as due to large cavities from mining activities which are close to water bearing surrounding rock and not have been backfilled for several decades safe enclosure is not taken into account as the integrity of the geological barrier is not guaranteed over time, a fact that was unknown when radioactive waste was emplaced into the Morsleben repository and the Asse mine. In case of the Gorleben site virgin salt rock is being explorated for repository purposes. When taking into account safety design aspects during planning of the repository, safe enclosure is achieved by an intact geological barrier acting in parallel with reliable geotechnical barriers.

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UNDERSTANDING THE EVOLUTION OF THE REPOSITORY AND THE OLKILUOTO SITE

K. Koskinen[1], Barbara Pastina[2]
[1]Posiva Oy, [2]Saanio & Riekkola Oy; Finland

Abstract

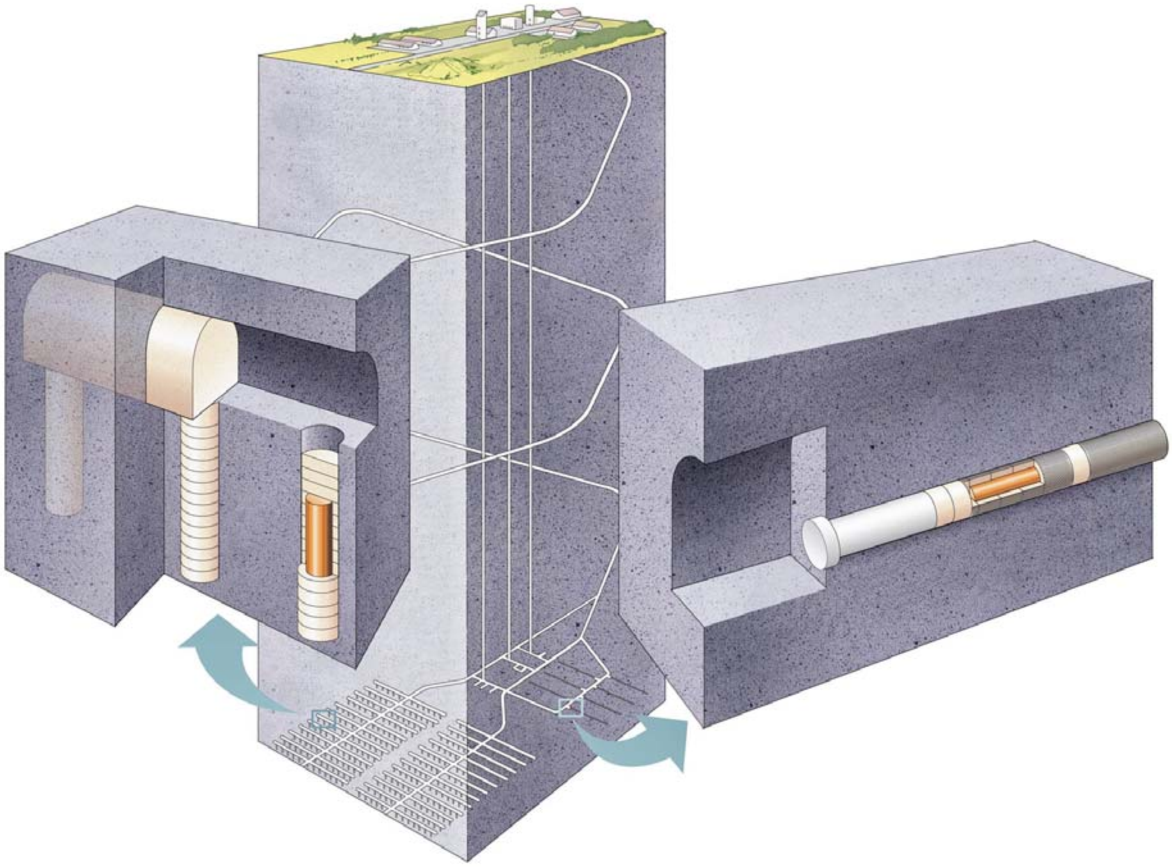
Posiva's Safety Case is organised in a portfolio including ten main reports: Site, Spent Fuel Characteristics and Inventories, Canister Design, Repository Design, Process, Evolution of the Repository and the Site, Biosphere Assessment, Radionuclide Transport, Complementary Evaluations of Safety, and Summary. This portfolio constitutes the basis of the Preliminary Safety Assessment Report, which will be presented to the authorities in 2012 as part of the repository construction license application. The Evolution report [1], which is the focus of this paper, is the main advance in the Safety Case portfolio since the implementation of the Safety Case plan [2] in 2005. The report provides the status of current knowledge with respect to the evolution of the site and the engineered barrier system and highlights areas where better understanding is needed.

Context

In May 2000, the Finnish Parliament promulgated a "Decision in Principle" to locate a spent fuel repository in Olkiluoto, on the Southwestern coast of the country. In 2009, Posiva will submit the first outline version of the Preliminary Safety Analysis Report (PSAR) in support of construction license application. The PSAR will then be gradually updated to become the actual licensing application, to be submitted in 2012. Posiva will submit its Safety Case at this time. A Final Safety Analysis Report (FSAR) will be submitted at the time of the operational license application, in 2018. The target is to begin disposal operations in 2020. Posiva is now in the site confirmation and repository design phase. In June 2004, Posiva began excavating the Olkiluoto Underground Rock Characterisation Facility, ONKALO, which may also be used as part of the future repository.

The Finnish spent fuel management is based on the KBS-3 concept for deep geologic disposal. This concept was originally developed by Sweden's Svensk Kärnbränslehantering AB (SKB). Figure 1 shows both the vertical (KBS-3V) and the horizontal (KBS-3H) options of the concept. Posiva's main design option is the KBS-3V but the development of the KBS-3H alternative continues in collaboration with SKB. A preliminary safety case for the KBS-3H alternative is expected in 2007. The KBS-3 concept aims at long-term isolation and containment of spent fuel assemblies in copper canisters with a nodular cast iron insert. The canister is emplaced several hundred metres deep into the bedrock. Each canister is isolated from the bedrock by a thick bentonite clay layer (the buffer). After emplacing individual canisters and bentonite buffer into deposition holes, repository tunnels and access routes to and from the surface are backfilled and sealed. Both the buffer and the backfill develop a swelling pressure due to their clay content as they saturate with water.

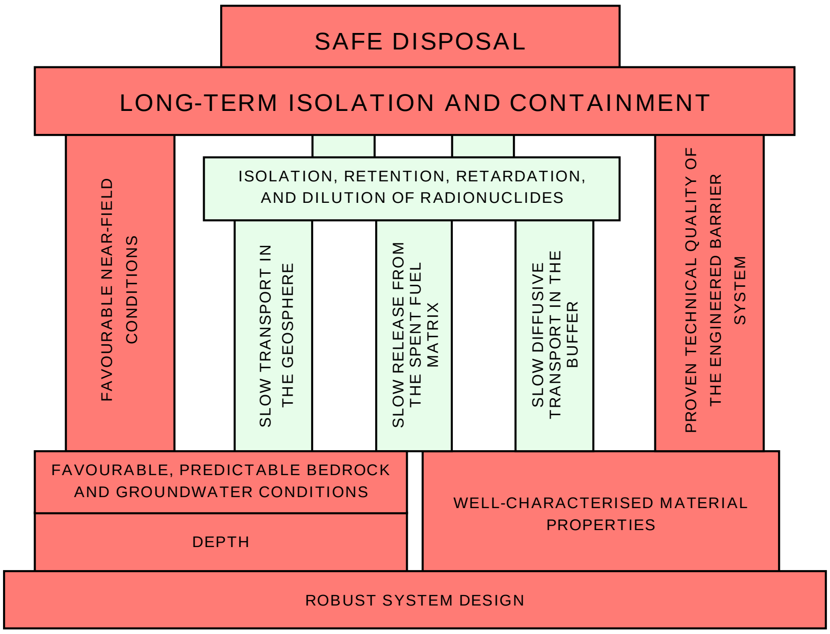
Figure 1. **The multi barrier KBS-3 concept showing both the vertical deposition (KBS-3V) concept on the left and the horizontal deposition (KBS-3H) concept on the right.**



Safety Concept

Posiva's safety concept for the KBS-3 design is shown in Figure 2. According to the safety concept, safety rests first and foremost on the ***long-term isolation and containment*** of radionuclides within the copper-iron canisters. Long-term containment within the canisters in turn depends primarily on the ***proven technical quality of the engineered barrier system*** (EBS) and ***favourable near-field conditions*** to promote their longevity and to ensure the functionality of the EBS.

Figure 2. **Outline of safety concept for a KBS-3 type repository for spent fuel in a crystalline bedrock (adapted from [1]). The safety concept is based on a robust system design. Dark grey pillars link primary safety features of the disposal system on which they primarily depend on to the overall goal of the safety concept (safe disposal). Light gray pillars indicate secondary safety features that may become important in the case of a radionuclide release from the canister.**



The technical quality of the EBS is favoured by the use of components with ***well-characterised material properties*** and by the development of appropriate acceptance specifications and design criteria. The site characterisation and repository design strategies at the Olkiluoto site are focused on a volume of bedrock situated between -400 and -700 metres below the surface where ***favourable and predictable bedrock and groundwater conditions*** are expected to be found, as required by Finnish regulations on spent fuel disposal [3]. The repository ***depth*** requirement from [3], aiming at decoupling the repository from the impacts of above-ground events, actions and environmental changes and at minimising the risks of inadvertent human intrusion, is satisfied in the current design.

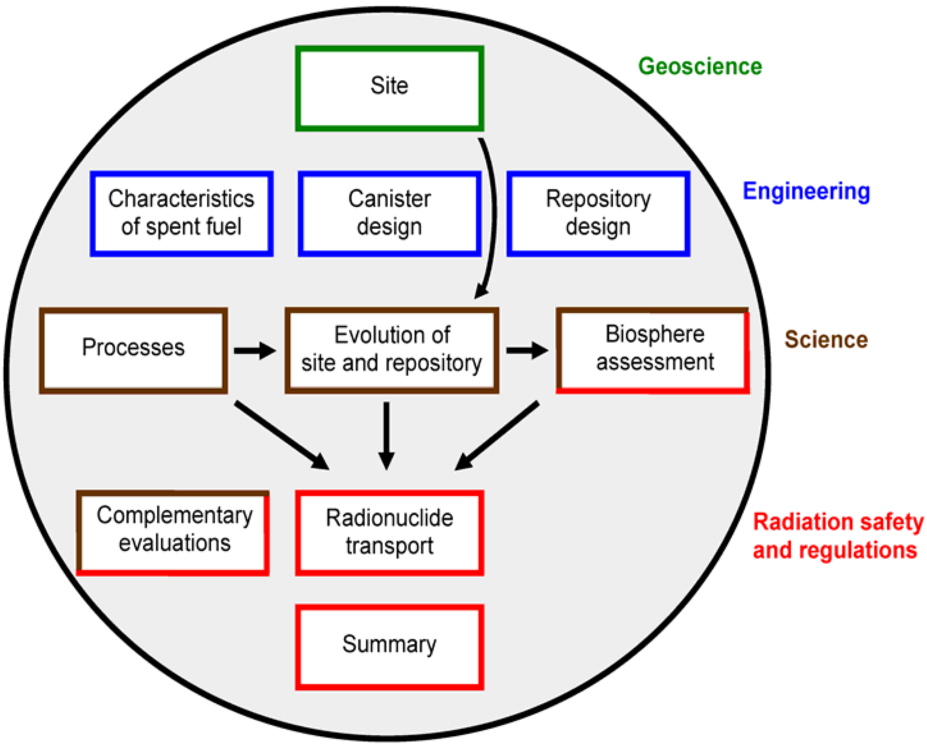
According to the requirement on multiple barriers [3], additional physical and chemical barriers form a secondary set of safety pillars (in light grey in Figure 2). These physical and chemical barriers become important in case the primary engineered barriers fail. They ensure that any radionuclides released from a failed canister undergo ***retention, retardation and dilution***. Retention, retardation and dilution of radionuclides are achieved with ***a slow radionuclide transport in the geospher***e, ***slow release from the spent fuel matrix, and slow diffusive transport in the buffer***. The safety concept relies on a ***robust system design***. A robust system is a system that is insensitive to imperfections in implementation and to uncertainties and assumptions used in long-term safety assessments.

Assessment basis

Posiva's safety assessment is based on the KBS-3 concept for repository design and EBS [4], the site (Olkiluoto) and the safety concept (Figure 2) that joins them. The purpose of the safety assessment is to assess the safety of the disposal concept under several possible courses of events and to discuss the importance of the remaining uncertainties. The safety assessment is linked to the assessment of the robustness of the repository design. A generic strategy for approaching this target is to use the following steps, also reported in [5 and 6]. First, the characteristics and state of the site and disposal facilities are described. Second, an assessment of the evolution of the site and the disposal facilities is provided with several assumptions of the main driving forces, such as climatic events. Third, estimates of long-term containment of radionuclides within repository facilities, the consequent concentrations of radionuclides in the environmental media and their radiological impacts are produced. The third step involves variants in details in implementation (defective implementation) and in unexpected external conditions. Forth, the compliance of the outcomes with regulatory requirements is evaluated. Posiva accomplishes this assessment strategy using a set of ten living and fairly independent reports, represented by boxes in Figure 3. These reports compose the “safety case portfolio”.

The figure shows how the safety case portfolio is managed to obtain input data to be used in the various safety assessment calculations. The Fuel, Repository Design and Canister reports are engineering-driven and contribute mostly to the data to be used in the safety assessment. The Site, Process, Evolution and most of Biosphere Assessment reports are science-driven and contribute mostly to the concepts (conceptual models) of the safety assessment and to a minor extent also to the data. The Radionuclide Transport, Complementary Evaluations and some of the Biosphere Assessment reports address radiation safety and regulatory requirements. The Summary report draws together the key findings and arguments and concludes with a statement of confidence in the long-term safety of the waste disposal programme. These reports are updated in average every three years or more often, if needed. In 2006, Posiva completed a cycle of information gathering (see “Path Forward” below) for the assessment basis and produced the Evolution report [1] as part of the analyses of "evidences and uncertainties". The following describes the main conclusions from this report.

Figure 3. **Main reports in the Safety Case portfolio. The colours of the boxes indicate the nature of the reports (geoscience, engineering, science, radiation safety and regulations). The arrows show the most important transfers of knowledge and data [2].**



Expected evolution of the repository and the site

The purpose of the Evolution report is to describe the expected evolution of the repository and the Olkiluoto site in terms of thermal, hydraulic, mechanical and chemical processes for each of the repository system components: geosphere, buffer, backfill and seals, and canisters. The time frames considered in the Evolution report are the following:

- Operational phase (up to approximately 100 years after first canister emplacement).
- Post-closure temperate phase (approximately 13 000 years AP and 170 000 years AP depending on the climate scenarios considered).
- From the onset of the next glaciation until the far future (corresponding to the end of the next glacial cycle, 125 000 years AP or 450 000 years AP, depending on the climate scenarios considered).

The Evolution report describes the evolution of the site under two climate scenarios: the Weichselian-R and the Emissions-M scenarios. The Weichselian-R scenario is based on the repetition of the latest (Weichselian) glacial cycle. In this scenario, the onset of the next glaciation will occur approximately 13 000 years after present (AP) and the entire cycle will end approximately 125 000 years AP, after which the climate will return to a situation similar to the current one. This scenario is based on the current knowledge of the evolution of the surface temperatures from the Greenland Ice Core Project [7].

The Emissions-M (M for moderate) scenario takes into account anthropogenic greenhouse gas emissions (especially CO_2). Such emissions are assumed to be neither extremely low nor extremely high but they lead to a longer temperate period before the onset of the glaciation (approximately 175 000 years AP). This scenario extends up to 450 000 years after present. The Emissions-M scenario is based on the climate scenarios considered in the European Community project BIOCLIM [8].

Understanding of the initial conditions of the repository and the site

Although the initial state of the canister, buffer and backfill could be described along with details on spent fuel inventory and repository design, the baseline hydrological and geochemical conditions will be disturbed to such an extent that baseline conditions of the bedrock needs to be re-evaluated with great care. In addition to this, the degradation of engineering materials, like cement, the transport of the related solutes with groundwater and the interaction potential of these solutes with the EBS in prospect need to be assessed.

Site-scale processes

Figure 4 summarises the main processes affecting the evolution of Olkiluoto site throughout the evolution phases considered. The figure also shows the evolution of the decay heat from the entire spent fuel inventory (6 000 tonnes of uranium) expected at repository closure. During the early evolution after closure, the main driving forces are the hydraulic gradients caused by repository construction and operation as well as the thermal and hydraulic gradients generated by the spent fuel decay heat. These gradients gradually disappear a few to a few tens of thousand years after repository closure.

Another source of hydraulic gradient, albeit much weaker, is the apparent land uplift (6-8 mm per year) as the earth crust at Olkiluoto is still recovering from the latest glaciation. Figure 4 shows that the relative shoreline level increases first due to ongoing land uplift and subsequently decreases during the glacial phase due to the depression (a few hundred meters) caused by the presence of a 2-km thick ice layer over the site. The repository level (-420 m) remains at the same in the figures for clarity.

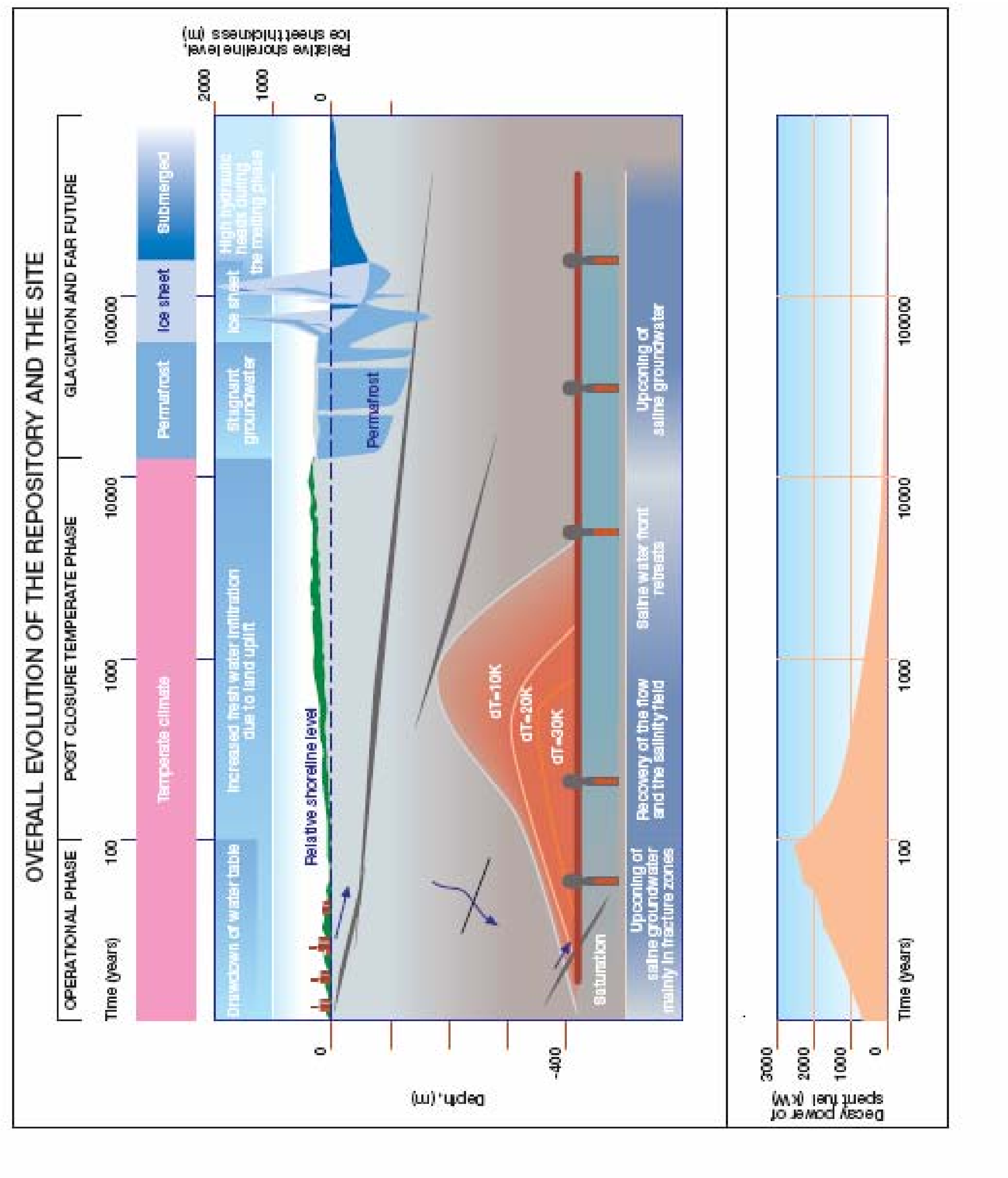
The onset of the next glaciation begins with the formation of a layer of permafrost according the Weichselian-R climatic scenario. Numerical simulations suggest that this permafrost layer will be 180 metres deep at most, remaining well above the repository level. The permafrost period is followed by short periods of warmer climate preceding the formation of an ice sheet. High hydraulic gradients due to conditions at the surface are expected during the ice-melting period. Post-glacial earthquakes are also potentially following the retreat of an ice sheet, although there are no signs of faults that have been active during the latest post-glacial period. The consequences of post-glacial earthquakes and rock shear movements on the canister are discussed below.

In the Weichselian-R climatic scenario the Olkiluoto site will be submerged throughout the glacial retreat phase, and the ensuing interglacial. No natural gradient or topographical feature giving rise to elevated groundwater flows rates is present. The density of water, which depends on the salinity among other things, is the driving force for seawater infiltration into the bedrock. Seawater salinities are not expected to be elevated due to the diluted glacial melt water still present in the area.

Marine water infiltration during a submerged stage at Olkiluoto to repository depths seems therefore unlikely but cannot be ruled out. In the event of marine water infiltration, higher sulphate and chloride concentrations could be present at repository depth.

As the climate continues to warm up and the land rises due to the post-glacial land uplift, hydraulic gradients associated with uncovering topographical undulation strengthen causing groundwater flows once again. According to the Weichselian-R scenario, the Olkiluoto site is expected to emerge from the water in about 125 000 years, and the glacial cycle would repeat again starting with a period of temperate climate similar to today’s conditions.

Figure 4. **Overview of the main processes considered during the expected evolution of the repository and site under the Weichselian-R climatic scenario. All canisters are assumed initially intact. The repository depth is –420 m in this figure according to the current design. The blue arrows note groundwater flow, black lines and dark grey areas mark fractures and deformation zones. The evolution of the decay power with time refers to the entire spent fuel inventory emplaced in the repository and the equipotentials of the temperature increase (dT = 10K, 20K and 30K) in the rock above the repository are shown as a function of time and depth. The evolution of the decay power with time refers to the entire spent fuel inventory (6000 tU) emplaced in the repository. The upper figure depicts topographic and geologic features while the horizontal axis represents time. Some features (e.g. fractures) are exaggerated in the figures for clarity.**



Canister-scale processes

The canister plays a key role in the safety concept of the KBS-3V repository. The goal is to maintain canister integrity and favourable conditions for the other components of the engineered barrier system (i.e. buffer and backfill) as their effectiveness also impacts the canister durability. Therefore, the main processes in the near field affecting the effectiveness of the buffer and backfill are discussed before the evolution of the canister itself.

The thermal and hydraulic gradients caused by construction and repository operation also create salinity gradients at repository level and affect the geochemistry at the canister scale. Groundwater salinity increases decrease the swelling pressure of the tunnels backfill. The long-term safety implication of a reduced swelling pressure in the tunnel backfill is the possible increase of the hydraulic conductivity in the near field. Higher hydraulic conductivity (compared to that of the surrounding rock) increases the transport of groundwater and solutes to and from the deposition hole so that the effectiveness of the EBS could be compromised and the canister lifetime reduced. Salinity changes do not affect the buffer swelling pressure because of its high density at deposition time.

The main processes occurring at the surface potentially affecting the salinity of the groundwater in the near field are the following (Figure 4): drawdown of the water table during the operational phase due to the open excavations, increased fresh water infiltration due to the land uplift in the early post-closure period, a stagnant flow or possible upconing of saline groundwater during the permafrost and in the case of a cold-based ice sheet and enhanced groundwater flow with possible upconing and diluted water intrusion during the glacial melting period. A submerged period follows the melting of the ice sheet and, during that period, intrusion of seawater (fresh/brackish) may take place.

As discussed above, the canister evolution is mostly affected by copper corrosion rates, which depend on the geochemical conditions in the near field and by rock shear movements, which may affect the mechanical integrity of the canister. The most important factors affecting the copper corrosion rate of copper are the rate of migration of ions (in particular chloride and sulphide) to and from the canister surface and the amount of available oxygen. Uniform corrosion and microbially induced corrosion are the most probable corrosion mechanisms during the operational phase due to the presence of high temperatures, oxygen, microbes, and stray materials introduced during repository construction. The oxygen trapped in the pores of bentonite at the beginning of operations is likely the primary source oxygen. Sulphate, along with methane and sulphate-reducing bacteria contribute to uniform corrosion by producing sulphide ions. The low concentrations of both ammonia and acetate in the repository and the low corrosion potential values make stress corrosion cracking an unlikely process [9].

Cementitious materials used for construction and groundwater control may interact with the canister. The highly alkaline water (pH > 12-13) resulting from cement porewater leachates reduces canister corrosion rates as the surface is passivated. However, passivated surfaces may become more prone to localised corrosion, especially in saline groundwater. The use of low pH cement (pH < 11) is currently considered for the areas closer to the canisters.

In saline conditions, general corrosion of copper occurs as long as oxygen and metallic copper are present. The uniform corrosion induced by chloride decreases the probability of local corrosion. In anoxic conditions, the susceptibility to chloride corrosion is low, the risk increases somewhat at low pH and high temperature (pH < 6, T > 80°C). Conditions for copper corrosion due to chloride in oxygen-free (anoxic) conditions are not likely but cannot be totally excluded.

Canister corrosion proceeds during the early post-closure period via general corrosion as long as oxygen is present. Uneven buffer swelling can also cause localised corrosion where the copper is in contact with the bentonite. The simultaneous presence of copper, residual oxygen (trapped in the bentonite pores), and chloride ions (in water and in the bentonite double layers) is favourable to pitting corrosion. Corrosion rates slow down when oxygen has been consumed and the environment has become reducing. Microbially induced corrosion is also possible but microbial activity decreases as the swelling pressure in the buffer increases. Large-scale tests in Äspö have shown that several corrosion mechanisms are active under unsaturated conditions. Uneven buffer swelling causes mechanical loads to the canister and deformation of the copper shell. However, experimental results show that these load do not threaten the integrity of the canister.

Copper corrosion rates are expected to decrease during the post-closure saturated period. The oxygen initially present in the near field has been depleted and the canister environment is now reducing. A fully saturated buffer effectively opposes groundwater circulation around the canister, impeding solute transport to and from the canister surface. Microbially induced corrosion is not viable because of the high swelling pressure in the buffer. Therefore, uniform corrosion via sulphide ions is the only viable copper corrosion mechanism during this period.

Uniform corrosion is also the only viable copper corrosion mechanism during the permafrost period. However, corrosion rates are expected to be extremely low due to the limited groundwater flow available to transport potentially detrimental species (e.g. oxygen, sulphide, chloride ions) to the canister. The methane thought to be present in the deeper layers of the bedrock may also contribute marginally to the microbial production of sulphide in the canister environment. In the presence of a cold-based ice sheet, low copper corrosion rates are also expected because of the weak hydraulic gradients.

The glacial melting periods (two are expected in the next glaciation, according to the Weichselian-R scenario) may enhance groundwater flows because of the pressure differential between the areas covered and those uncovered by ice. Upconing of deep saline groundwater is possible due to these pressure differentials. Furthermore, intrusion of diluted glacial melt water at repository depth cannot be ruled out. Glacial melt water, because of its low ionic strength, could cause chemical erosion of the buffer and the backfill and enhance canister corrosion rates. Canister corrosion rates could also be further enhanced by the presence of oxygen dissolved in the glacial melt water. In addition to the potential loss of buffer due to glacial melt water intrusion, the upconing of dissolved gases from the deeper layers of the bedrock could enhance canister corrosion but only if sulphate and sulphate-reducing bacteria are present. Microbially induced corrosion can be possible if oxygen and organic materials are introduced at repository depth from glacial melt water intrusion. The likelihood, extent and consequences of glacial melt water intrusion are currently being evaluated. Post-glacial earthquakes following the retreat of an ice sheet cannot be ruled out at this time although no signs of past post-glacial earthquakes have been observed at the Olkiluoto site. The consequences of post-glacial earthquakes and rock shear movements on the canister estimated in the Evolution report is that one canister (out of 3 000 in the repository) every 240 000 years is susceptible to fail due to a rock shear movement.

In the event of marine water infiltration during a submerged stage at Olkiluoto at repository depths, higher sulphate and chloride concentrations could be present in the vicinity of the canister. Sulphate ions may enhance corrosion rates in the presence of sulphate-reducing bacteria, while chloride ions may induce local corrosion. The chloride concentration is not expected to be high because of the glacial origin of these waters. The concentration of chloride ions needed to induce corrosion of the copper canister is much higher (over 55 g/L TDS).

At the end of the evolution period studied under the Weichselian-R climate scenario (approximately 125 000 years AP), the canisters are still expected to be intact (except if a large rock shear movement crosses a section of the repository or in the presence of initial penetrating defects). After approximately 250 000 years, the activity remaining in the fuel is similar to that of a large uranium ore body. The repository can then be imagined as a large uranium ore body well isolated from surface waters. Experience indicates that such ore bodies do not pose a surface hazard. Except for defective canisters or those breached by a large rock shear movement, the canisters are expected to last over one million years based on the corrosion rates expected during the evolution time frame.

Although it is expected that most of the canisters will be intact at emplacement time, the evolution report also describes the evolution of a defective canister with a non-penetrating defect and a defective canister with a penetrating defect. The evolution of a defective canister both with an initially non-penetrating defect is similar to that of an intact canister. Over a sufficiently long time frame, all canisters will eventually fail. If defects are present, these will determine the most likely location for failure by corrosion, since these are where copper coverage is thinnest. Except for defective canisters or those breached by a large rock shear movement (1 canister every 240 000 years), the canisters are expected to last over one million years. The case of 3 initially penetrated canisters (out of 3 000) was also considered but no radionuclide calculation data were carried out in this Evolution report. Even if the defect is located near the base of the canister, calculations show that following the water inflow from the hole, no water will be expelled because of the low pressure buildup inside the canister. A previous calculation of doses released from 1/1 000 initially penetrated canisters showed that they remained within the regulatory release limits [10].

The evolution of the repository and the site under the Emissions-M climate scenario is similar to that for the Weichselian-R scenario except the timing of climatic changes (permafrost, ice sheet, glacial melting) and the effects of the prolonged surface water infiltration due to the continuing land uplift. During the post-closure temperate phase, surface waters are pushed deeper and deeper in the bedrock with time. Surface waters that penetrate to repository depth will be subject to chemical interactions with the rock, increasing their ionic strength. Therefore, chemical erosion of bentonite is not expected in the temperate period of the Emissions-M scenario. It is assumed that the climate evolutions in the Emissions-M scenario will be similar to the Weichselian-R from the development of the ice sheet onwards. Canister corrosion under conditions similar to those during the post-closure temperate phase will continue for over 160 000 years compared to the Weichselian-R scenario, until the onset of the next glaciation (175 000 years AP). Canister corrosion rates are of the order of hundreds of micrometers at most during the post-closure phase. From the onset of the next glaciation, the same processes as in Weichselian-R scenario will occur but the consequences on canister durability are less relevant because the permafrost depth and thickness of the ice layer are much smaller than in the case of the Weichselian-R scenario and because the spent fuel inventory would have decayed to even lower levels by the onset of the next glaciation.

Path forward

The Evolution report shows that the operational and the early post-closure phase are the most eventful periods for the geosphere and the canister. The thermal effects on the host rock from the spent fuel decay heat remain significant for several hundred years after repository closure. Thermal gradients also accelerate geochemical reaction, such as oxygen consumption at repository depth, reactions with engineering and stray materials (introduced during the construction of the repository) and microbial activity.

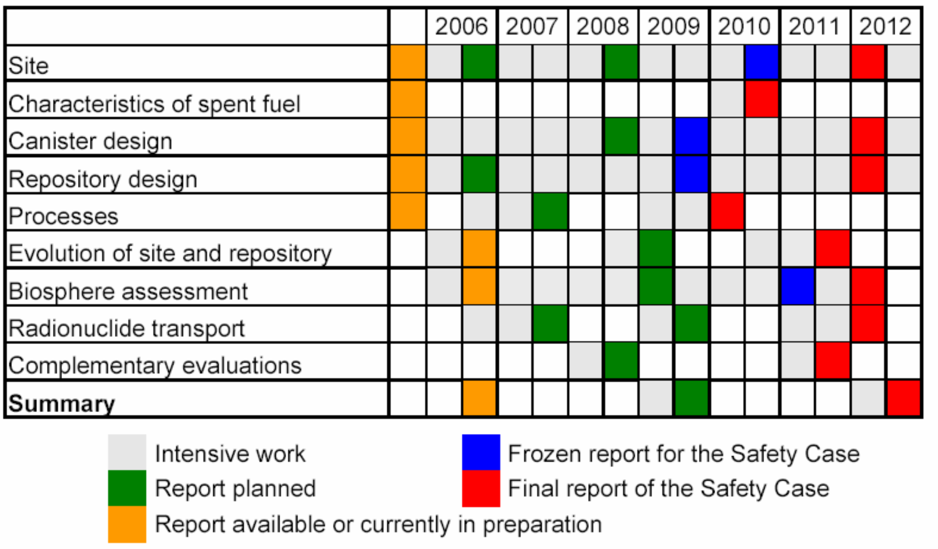
The remaining of the post-closure phase and the beginning of the glacial phase are uneventful from the canister point of view because of the low hydraulic and temperature gradients expected. The glacial melting phase will cause high hydraulic gradients again and possibly glacial melt water intrusion, which are potentially detrimental to canister durability.

The canister durability depends on the thermal, hydraulic, mechanical and chemical conditions in the deposition hole. These conditions vary considerably not only with time but also from deposition hole to deposition hole depending on the local salinity levels, on the timing and sequence of canister deposition and on the degree of saturation of the buffer and backfill which in turn affect the local characteristics of the sparsely fractured zone where a deposition hole is located. Therefore, although detrimental conditions to canister durability may be found locally, these are not expected to affect all the canisters at once. In other words, no event leading to the simultaneous release of the entire spent fuel inventory is expected before one million years.

The main open issues identified in this Evolution report warranting further attention are the bentonite saturation process and the interaction of the engineering and other residual materials with the EBS. The role of the EDZ and possible thermal spalling on the hydraulic conductivity in the near field is also currently not well understood. These issues are relevant to the early evolution period. The evolution of other system components that are not part of the engineering barrier system (e.g. repository plugs and seals) is important in the latter part of the post-closure phase and beyond because it may affect the effectiveness of the engineered barriers and the isolation of the repository from the surface. Plugging and sealing systems are assumed to provide a reasonable isolation of the repository system from the surface as long as this is needed for safety reasons, but the basis of this assumption will be evaluated in the coming years. During the glacial melting period, the extent and consequences of a diluted (and possibly oxygenated) water intrusion will also be further analysed.

To improve the understanding of the evolution of the site and repository, a new cycle of information gathering has begun. Plans for research, technical design and repository development activities are described in [5]. The schedule of the Safety Case portfolio until the construction license application (2012) is presented in Figure 5. An interim update of the status of the safety case is scheduled for 2009. Frozen versions of the site, canister design, repository design and biosphere assessment reports will be ready between 2010 and mid-2012 so that the Summary Safety Case report can be compiled by the end of 2012.

Figure 8. **Overall schedule for the main reports of the Safety Case (updated from [1]).**



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MAKING THE POST-CLOSURE SAFETY CASE FOR THE PROPOSED YUCCA MOUNTAIN REPOSITORY

P. Swift[1], A. van Luik[2]
[1]Sandia National Laboratories, [2]United States Department of Energy; USA

Introduction

The International Atomic Energy Agency (IAEA), in its advisory standard for geological repositories promulgated jointly with the Nuclear Energy Agency (NEA) of the Organisation for Economic Co-operation and Development, explicitly distinguishes between the concepts of a safety case and a safety assessment. As defined in the advisory standard, the safety case is a broader set of arguments that provide confidence and substantiate the formal analyses of system safety made through the process of safety assessment. [1]:

Definitions of safety assessment and the safety case

Safety assessment *is the process of systematically analysing the hazards associated with the facility and the ability of the site and the design of the facility to provide for the safety functions and to meet technical requirements. . . .*

The safety case *substantiates the safety, and contributes to confidence in the safety, of the geological disposal facility. The safety case is an essential input to all the important decisions concerning the facility. It includes the output of safety assessments . . . , together with additional information, including supporting evidence and reasoning on the robustness and reliability of the facility, its design, the design logic, and the quality of safety assessments and underlying assumptions. . . .*

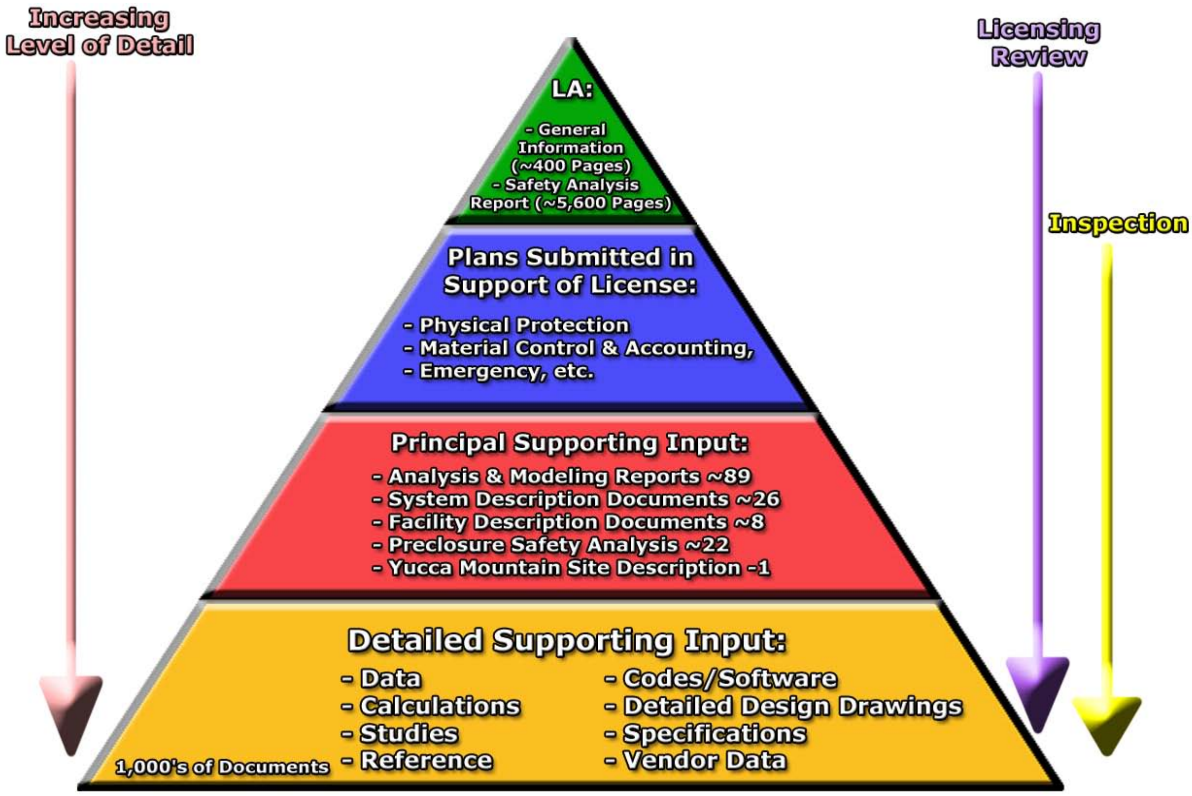
Although the IAEA's definitions include both pre-closure (i.e. operational) safety and post-closure performance in the overall safety assessment and safety case, the emphasis in this paper is on long-term performance after waste has been emplaced and the repository has been closed. This distinction between pre- and post-closure aspects of the repository is consistent with the U.S. regulatory framework defined by the U.S. Environmental Protection Agency (Chapter 40 of the Code of Federal Regulations, Part 197, or 40 CFR 197) [2] and implemented by the U.S. Nuclear Regulatory Commission (Chapter 10 of the Code of Federal Regulations, Part 63, or 10 CFR 63) [3]. The separation of the pre- and post-closure safety cases is also consistent with the way in which the U.S. Department of Energy has assigned responsibilities for developing the safety case.

Bechtel SAIC Company is the Management and Operating contractor responsible for the design and operation of the Yucca Mountain facility and is therefore responsible for the preparation of the pre-closure aspects of the safety case. Sandia National Laboratories has lead responsibility for scientific work evaluating post-closure performance, and therefore is responsible for developing the post-closure aspects of the safety case. In the context of the IAEA definitions, both pre-closure and post-closure safety, including safety assessment and the safety case, will be documented in the license application being prepared for the proposed Yucca Mountain repository, and in the documents that support that license application.

Organisation of the Yucca Mountain license application

The Yucca Mountain license application now under development is illustrated in terms of its place in a document hierarchy in Figure 1. The license application is the green pyramid-top. In turn it is underlain by plans and reports and data, including over a hundred major documents and literally thousands of supporting documents. Consistent with the expectations of the U.S. Nuclear Regulatory Commission, the case for safety is made in the license application, referring to the supporting scientific and technical documentation as needed.

Figure 1. **The license application document hierarchy**



The license application will consist of two sections, one giving general information, and one giving a report on safety analyses for both the operational and post-closure phases (Table 1).

Table 1. **Content of the two major parts of a license application for the proposed Yucca Mountain repository**

- General Information (GI):
  - General Description.
  - Proposed Schedules for Construction, Receipt and Emplacement of Waste.
  - Physical Protection Plan.
  - Material Control and Accounting Programme.
  - Site Characterisation.

- Safety Analysis Report (SAR):
  - Repository Safety Before Permanent Closure.
  - Repository Safety After Permanent Closure.
  - Research and Development Programme to Resolve Safety Questions.
  - Performance Confirmation Programme.
  - Administrative and Programmatic Requirements.

The safety case consists mainly of the items in Table 1 labeled "Repository Safety Before Permanent Closure," and "Repository Safety After Permanent Closure." Elements of these two sections that contribute to the overall safety case are summarised in Table 2, along with institutional aspects of the programme that provide confidence in the implementation of the technical programmes.

Table 2. **Elements of the safety case**

- Pre-closure Safety Case (the case for operational safety):
  - Pre-closure safety analysis – event sequences categorised by frequency.
  - Safety margin and defense in-depth.
  - Analysis of Category 1 & 2 event sequences.
  - Industry precedent and experience.
  - Technical specifications and surveillance.

- Post-closure Safety Case (the case for passive safety after final closure):
  - Total system performance assessment (TSPA).
  - Identification and description of multiple barriers.
  - Analysis of potentially disruptive events.
  - Insights from natural analogues.
  - Performance confirmation.

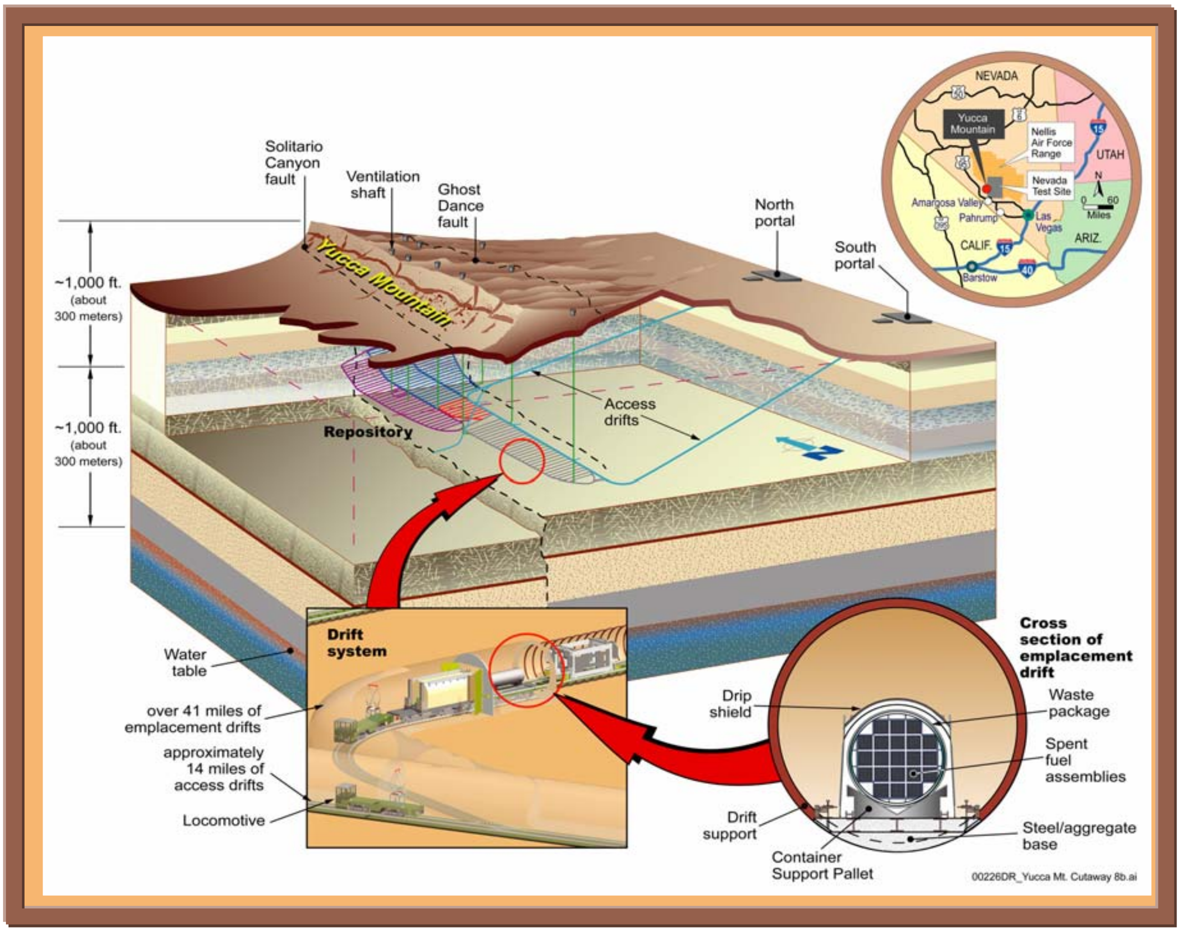
- Institutional Assurance (the case for an institutional environment that provides confidence in the technical bases for the safety case:
  - Quality Assurance.
  - Safety Conscious Work Environment.

The focus of this paper is on the post-closure case for repository safety.

Overview of the technical basis for post-closure performance

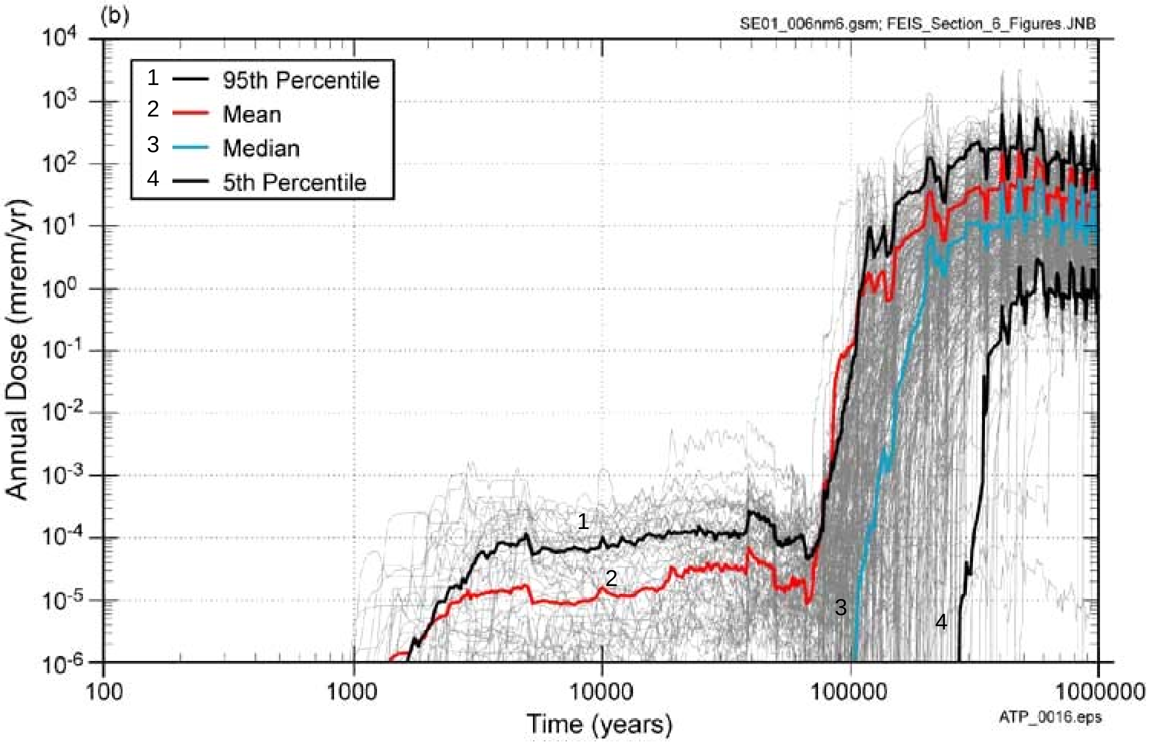
The proposed Yucca Mountain repository will be placed into an unsaturated volcanic mountain ridge, about halfway between the surface and the water table (Figure 2). Post-closure safety is dependent on the characteristics of the unsaturated zone through time (e.g. flow and contaminant transport as climates change), on the characteristics of the engineered system within that unsaturated zone (e.g. resistance to corrosion and physical damage and waste form dissolution behaviour over time, and stability of the mined openings as seismic activity occurs), and on the characteristics of the saturated zone over time (elevation of water table, flow and contaminant transport to the accessible environment).

Figure 2. **The proposed Yucca Mountain repository in its unsaturated-zone setting (inserts illustrate general location, the engineered system in a drift, and a conceptual drawing of the automated emplacement scheme).**



The quantitative aspects of the post-closure safety case will be based on computer modeling of the potential evolution of the system over time. These models will account for the uncertainty that is unavoidably present in estimates of the future behaviour of natural and engineered systems through a Monte Carlo approach in which multiple simulations will be performed using sampled values of uncertain model inputs. Model results will be displayed as a set of estimates of annual dose to a hypothetical individual, whose lifestyle and characteristics are prescribed by regulation. Results will be shown with a mean, median, 5th and 95th percentile outcome denoted, to provide decision makers with a clear representation of the uncertainty in modeled performance. Figure 3 provides an illustration from a past total-system performance assessment. The multiple grey curves illustrate the uncertainty in the results.

Figure 3. **illustration of previous [4] nominal case safety evaluation for the proposed Yucca Mountain repository (300 Monte Carlo realisations, no disruptive events, ICRP 30 dose model). As noted in the document in which these results were originally published [4] the absolute value in these outcomes may be divided by 4, approximately, if one uses the internationally accepted ICRP 72 (International Commission on Radiological Protection Publication 72) [5] dose model rather than the older ICRP 30 [6,7,8] model used in these calculations.**



A logical question, seeing results of this nature, is why one should have confidence in decisions based on statistics applied to such an uncertain range of outcomes? The answer, in terms of confidence, lies in the observation that decisions can be based on the central tendency (e.g. the mean, median, or other measure) of the distribution, with consideration of the full range of uncertainty. Human decisions invariably must accommodate uncertainty, and sound decisions are best made with full consideration of the range of uncertainty. Choices about what values within the range of uncertainty to emphasise in decision making are fundamentally societal decisions, rather than scientific ones. Although a decision could, in principal, be based on any specified value from the range of outcomes, including extreme outliers, confidence in the reliability of the model results is greatest for values that correspond to stable statistical measures of the full distribution. The U.S. Environmental Protection Agency and the U.S. Nuclear Regulatory Commission acknowledge this observation through their requirements to regulate on the peak of the mean annual dose estimated for the system.

The U.S. National Academy of Sciences provides further support for the use of uncertain model results in decision making. In the context of a report on the conduct of probabilistic seismic hazard assessments [9], the Academy notes that decisions informed by calculations of this nature ought to be based on the central tendency of the distribution of outcomes, and on the robustness of that central tendency as new information becomes available with time. This expectation that the measure of central tendency be shown to be robust as new information becomes available is specifically acknowledged in U.S. Nuclear Regulatory Commission requirements for performance confirmation activities that continue scientific investigations of the repository system after construction and waste emplacement has begun. As required by regulation and as planned by the DOE, performance confirmation activities [10] will be designed to challenge basic data and assumptions underlying the safety assessment, allowing the DOE to confirm (or refute) the technical basis for the post-closure safety case during the operational period.

Additional confidence comes from objective demonstration that the quantitative estimates of performance have been developed following sound scientific processes including thorough analysis, documentation, and review. Relevant to achieving that goal, Table 3 lists representative conditions that will help support the conclusion that the safety evaluation is credible.

Table 3. **Conditions that support a finding that a safety evaluation is credible**

- The evaluation draws from a design and scientific data basis that is sufficient to support a meaningful total-system level evaluation.

- The evaluation uses calculational tools that have been independently reviewed and found to be competent in structure.

- Analyses that support the evaluation use the input data and exercise the calculational tools competently.

- The evaluation considers and explains uncertainties and other features of the calculational outcomes to demonstrate knowledge of the system and understanding of its behavior.

- In all of the above, the evaluation includes consideration of additional lines of evidence:
  - Data and information about comparable natural and technological systems
  - Comparisons in terms of structure and approach with comparable but independently created calculational tools
  - Comparisons with other models applied to the same system, or with the current model applied to different systems with selected analogous features and processes
  - Comparisons with, and explanations of differences in, previous analyses of the same system by the same organisation, reflecting known changes in calculational tools and in supporting design and scientific data.

Producing and documenting these arguments is a complex and time- and labor-intensive undertaking. Fortunately, for the proposed Yucca Mountain repository effort, there have been two independent safety evaluations with independently developed tools. One organisation also performing safety evaluations for a repository at this location is the regulator, the U.S. Nuclear Regulatory Commission (NRC) [11]. Another is the nuclear electric power industry, through its Electric Power Research Institute (EPRI) [12]. Understanding the differences in outcomes between the DOE, NRC and EPRI safety evaluations, in terms of tools, data, and assumptions, is a powerful additional line of evidence for having confidence in the DOE safety evaluations.

Independent technical reviews also can add confidence if properly responded to. Failure to respond to constructive criticism from independent reviewers, including taking substantive corrective actions where appropriate, would not lead to enhanced confidence.

System-level safety evaluations of a potential Yucca Mountain repository have been performed by DOE since the mid to late 1980s [4, for a recent example]. These analyses have been reviewed by NRC as part of a pre-licensing Key Technical Issue (KTI) [13] identification and resolution process. As previously noted, both NRC and EPRI have performed system-level analyses over this same time period [11,12].

All of the above analyses have been reviewed by technical oversight boards (US Nuclear Waste Technical Review Board or NWTRB, and the Advisory Committee on Nuclear Waste or ACNW). The DOE TSPA has been peer-reviewed in the past, including by Budnitz, Ewing, Moeller, Payer, Whipple and Witherspoon [14] and by the Organisation for Economic Co-operation and Development/Nuclear Energy Agency (OECD/NEA) and the International Atomic Energy Agency (IAEA) [15]. Some of the observations from the NEA/IAEA review that provide confidence regarding the reviewed system-level analysis are given in Table 4.

Table 4. **Observations and suggestions from the NEA/IAEA (2002) Peer Review of the DOE Total System Performance Assessment in support of the site recommendation**

- . . . the general approach to TSPA, and the USDOE approach of building on an iterative series of performance assessments, conform to international best practice. . . .
- . . . structure of the TSPA-SR methodology, and . . . [the] approach of building on an iterative series of performance assessments, conform to international best practice.
- The structured abstraction process linking process-level models to assessment models is at the forefront of international developments.
- . . . the FEP [Features, Events and Processes] methodology. . . [is] in agreement with international best practice . . .
- . . . places far greater emphasis on probabilistic assessment than equivalent programmes in other countries . .
- . . . does not emphasise natural analogues as much as in some other international studies.
- “While presenting room for improvement, the TSPA-SR methodology is soundly based and has been implemented in a competent manner.”

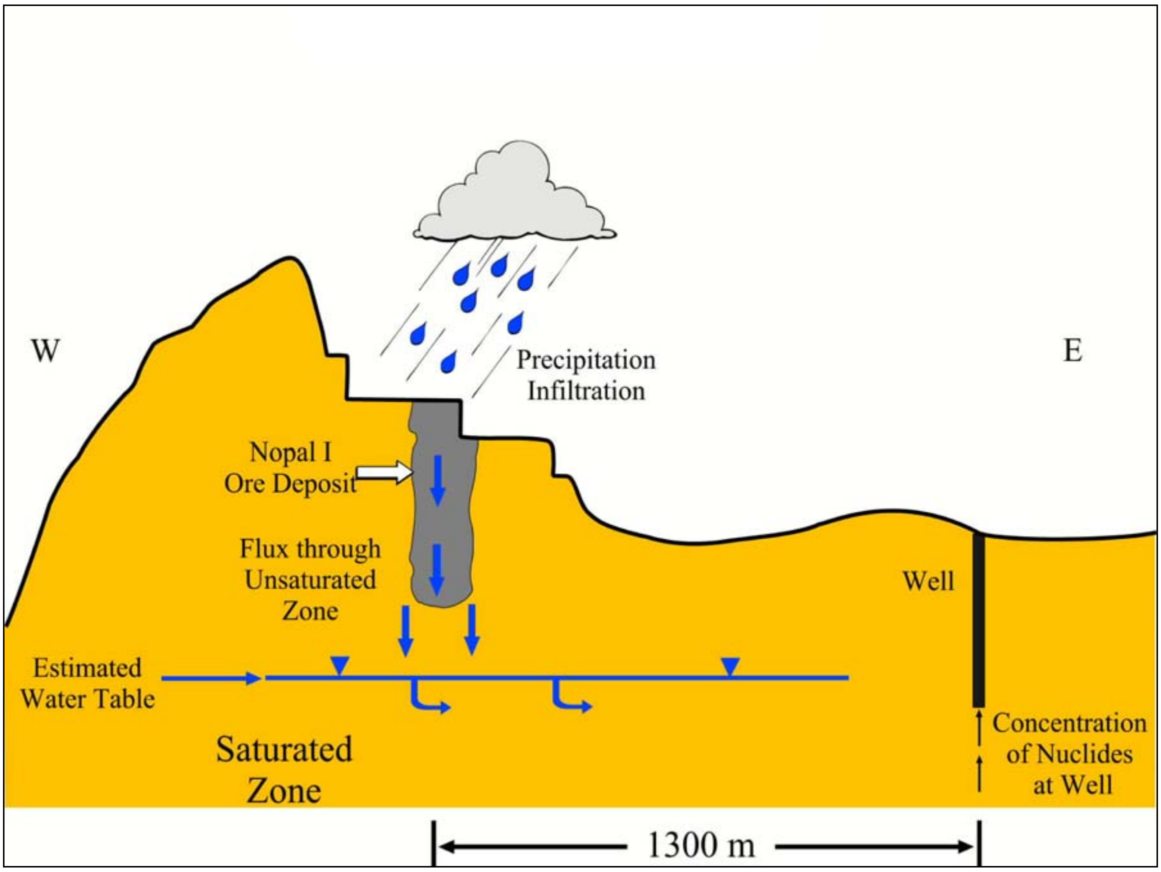
Several critical observations were also made in the NEA/IAEA review, suggesting a need to develop a more comprehensive safety case (the product reviewed was only a safety assessment), and a need to update the regional saturated zone flow model that provided boundary conditions to the site-scale flow (and thus determines radionuclide transport) model.

Natural analogues are generally seen as providing additional, independent lines of evidence for process behavior, and may be particularly useful for evaluating models of processes that span tens of thousands of years and are therefore not amenable to corroboration by direct observation [16]. Among the scientific documents providing a general level of support to the License Application effort is one that provides a synthesis of natural analogue work done and considered in the Yucca Mountain programme of work [17]. This document summarises information that has appeared in many analysis and modeling reports that provide supporting information to process-level models. Each technical document that directly supports the scientific content of the License Application will, where appropriate, cite specific aspects of the analogue work that gives insight, or otherwise supports, the data and assumptions fed into the safety evaluation.

An example of an analogue being studied by both the DOE and the NRC to provide insight, and hence build confidence in the Yucca Mountain safety assessment is the Nopal I uranium ore body and mine in the Sierra Peña Blanca, north of Chihuahua city in Mexico. The Nopal I analogue (Figure 5) is comparable to Yucca Mountain in a number of important ways.

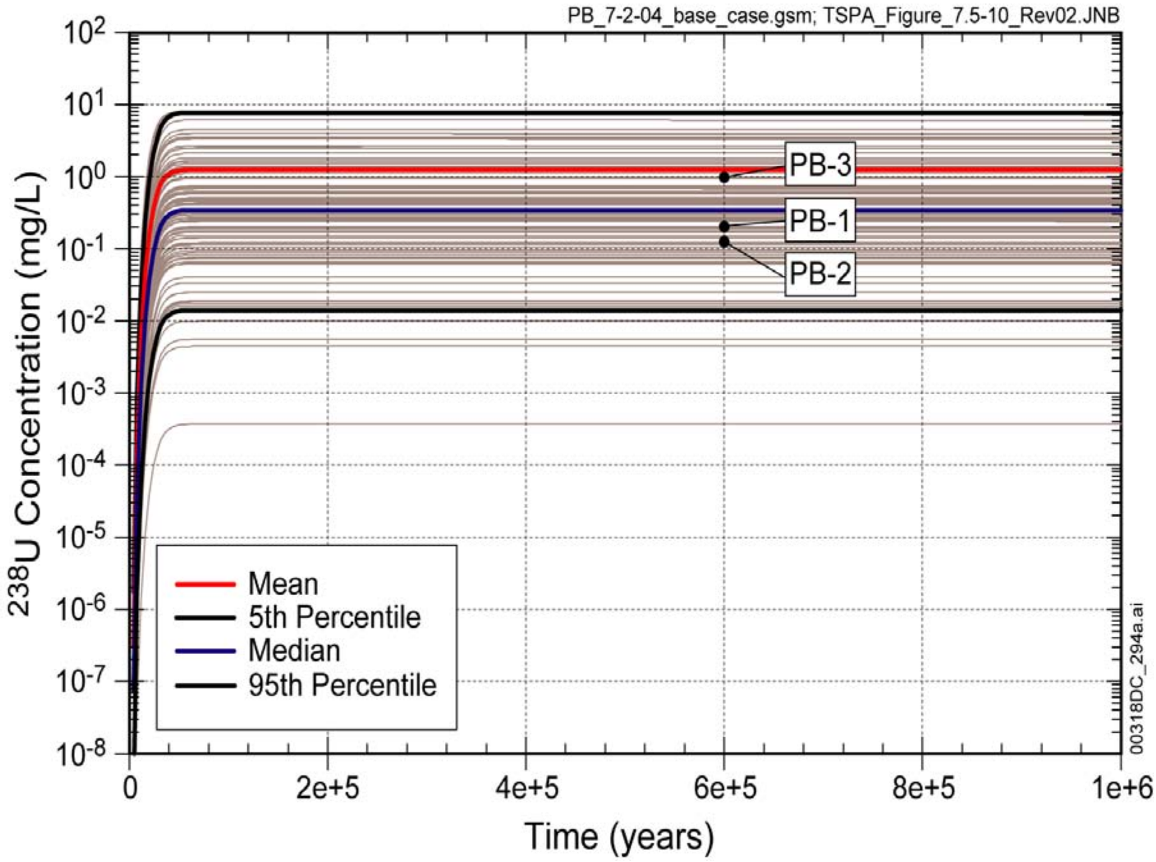
(1) Its UO_2 uranium ore deposit is analogous to spent nuclear fuel.
(2) Its fractured, welded, and altered rhyolitic ash flow tuffs overlie carbonate rocks.
(3) Its climate is semiarid to arid.
(4) Its geochemistry is oxidising, and has been for more than 3 million years.
(5) Its ore lies in the unsaturated zone above the water table.

Figure 5. **Conceptual Model for the Nopal I Uranium Deposit Implemented in the Peña Blanca Natural Analogue Model [18]**



The DOE has developed a Peña Blanca Natural Analogue Performance Assessment Model based on and simplified from its Yucca Mountain Total System Performance Assessment model [18]. Results of field investigations and laboratory analyses of rock and water samples from Nopal I were used to calibrate the Peña Blanca Natural Analogue Model, and the model was tested using Monte Carlo simulations of the mobilisation and transport of radiouclides from the ore to withdrawal wells downgradient in the saturated zone. Comparisons of model results with data from water samples are encouraging (Figure 6), and additional sampling from an ongoing programme of investigations in and around the ore body will help refine the model. The ultimate goal of this work is to lend confidence to the modeling approach being used for the comparable processes at the proposed Yucca Mountain repository.

Figure 6. **Base-case Simulation for ^{238}U for 100 Realisations Compared with Observed Concentrations in three Saturated-Zone Boreholes near the Nopal I Uranium Deposit [18]**



Conclusions

Does all of the above taken together constitute a safety case? We believe that it does. As in times past, however, the DOE will also create less-technical documents to explain to audiences that are not composed of specialists what the basis is for our asking for a license to construct a repository. These less-technical documents will communicate why we believe there is a basis in our license application for the regulator to find, with sufficient confidence, that there is a reasonable expectation of safety should the DOE be allowed to build this repository.

In the spirit of confirming the stability of the primary performance measure of regulatory interest, scientific work will continue during construction and will inform the license amendment request to allow us to enter the operational phase. Thereafter, scientific work will continue to support any changes in operations or design during the decades that the repository will be loaded with waste. Prior to final closure, all changes in knowledge and design from these previous decades will be used to show that there is sufficient confidence in the passive safety of the system to allow it to be closed and sealed. Even then, however, there will be continued monitoring and protection of the site, using both passive and active means, as long as (future) society deems it necessary.

Finally, it should be noted that an essential component of this safety case, and perhaps of any safety case, is a high level of confidence that there are strong institutional processes in place to ensure that execution of the scientific work and its documentation is sound, and that potential problems with the site are identified and addressed fairly and openly. If there are significant doubts about the quality of the technical work or the openness and fairness with which it is presented, confidence in the safety case will be diminished. The creation and use of an effective Quality Assurance programme is vital to ensuring that technical work is sound and correctly documented. Similarly, the creation of an open environment in which those persons most knowledgeable about the project, i.e. the scientists and engineers engaged in evaluating the safety case themselves, are free to raise concerns will help ensure confidence that potential problems are not overlooked. If the proponent and its experts have doubts about system safety, confidence can not legitimately be expected in others. The NRC requires, and the DOE strongly endorses, the creation of a "safety conscious work environment" (SCWE) in which all participants in the project have the right and obligation to raise concerns potentially related to the safety (both operational and long-term) of the facility, without fear of retribution. Demonstrating that such an environment exists and works is a required part of the documentation supporting the License Application. This policy empowers the Yucca Mountain work force, at any level, to voice concerns and even to stop work if there is a legitimate safety issue. The effectiveness of this programme adds credibility to the declaration by DOE and its analysts that there is sufficient confidence in the safety case to allow progression to the next phase in the life of this repository project.

Acknowledgments

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ELEMENTS OF THE SAFETY CASE FOR THE MORSLEBEN REPOSITORY BASED ON PROBABILISTIC MODELLING

J. Wollrath
Bundesamt für Strahlenschutz (BfS), Germany

M. Niemeyer, G. Resele
Colenco Power Engineering AG, Switzerland

D-A. Becker, P. Hirsekorn
Gesellschaft für Anlagen- und Reaktorsicherheit (GRS), Germany

Abstract

The Morsleben nuclear waste repository (ERAM) for low- and intermediate-level mainly short-lived waste is located in a former salt mine. The closure concept was developed in parallel and interacting with the safety assessment. The safety concept is based on extensive backfilling with salt concrete complemented with seals between the main disposal areas and the rest of the mine building. Thus, the entire system exhibits a barrier effect through a partially redundant combination of several processes. However, in the formal safety assessment no credit is taken from the barrier effect of the extensive backfill.

In the safety assessments, the different possibilities of system development, the resulting array of potential fluid movement and a large number of potential radionuclide migration pathways are mapped in the bandwidth of derived parameters. The calculated response of the system to parameter variations is non-linear. Different processes may compete and compensate each other. Hence, the common practice to choose a "conservative" parameter set for the safety assessment is a priori impossible.

The safety assessment has been performed independently by two groups with different computer models, for the same closure concept and the same basic parameters but independent conceptual approaches. Both groups have performed deterministic and probabilistic dose calculations. The results match well; the differences can be explained on basis of the model approaches. Although a large bandwidth is considered for a number of parameters the maximum radiation exposure remains clearly below the applicable dose limit for nearly all calculations, demonstrating the robustness of the system. These aspects significantly contribute to confidence building in the Safety Case for ERAM.

Introduction

The Morsleben radioactive waste repository (ERAM) in Saxony-Anhalt, near Helmstedt, Germany is located in a former salt mine in the geological formation of Zechstein/Thuringium (Upper Permian). The salt formation consists of folded rock salt, potash salt and anhydrite stratifications. The overall thickness of the salt formation is between 350 m and 550 m and its top lies about 260 m below ground surface. Access to the mine is provided through two shafts: The 525 m deep Shaft

Bartensleben connects 4 main mining levels between 386 m and 596 m b.g.s., the 520 m deep Shaft Marie connects two main levels. Due to earlier rock salt and potash mining, many caverns exist in this mine with dimensions of up to 100 m in length, and 30 m in width and height. The total volume of mined structures amounts to about 8 000 000 m^3.

Emplacement of predominantly short-lived low-level radioactive waste in the Morsleben repository started in 1971. Different areas of the mine were used to dispose of the waste using different techniques such as dumping of solid waste, stacking of drums and cylindrical concrete containers, and in-situ solidification of liquid waste. By the end of the operational phase in September 1998 a total waste volume of about 36 800 m^3 with a total activity of approx. $1.7 \cdot 10^{14}$ Bq had been emplaced.

The licensing procedure for the closure of the repository has been initiated. The closure concept is based on extensive backfilling of the salt mine with an inexpensive concrete mixture (salt concrete) using state-of-the-art technology and on the sealing of the disposal areas. The objectives of safety assessments have been to optimise the measures for the closure concept and to demonstrate operational and long-term safety. The results are now part of the safety case of the repository.

Evolution of the closure concept

Considering the special situation of the Morsleben project – decommissioning of a repository in a former production mine, where no further waste emplacement takes place and where future inflow of brine from the cap rock can not be excluded –, the development of the technical concept for the closure of ERAM had to be a stepwise iterative approach. Several potential closure concepts were elaborated and analysed in parallel. They were assessed by different teams with respect to technical feasibility and long-term safety, the latter by preliminary safety analyses. One closure concept – the encapsulation concept – was excluded because the technical feasibility was doubtful, especially regarding the realisation of the most important technical barrier. Another concept, the pore reservoir concept, seemed to be feasible in principle but was discarded because it required sophisticated technical measures and its realisation bore risks that could only be excluded by a very detailed exploration of the host rock. The two remaining concepts both envisaged sealing of the major disposal areas and backfilling of open cavities; they differed by the amount of backfill material. These two concepts were less ambitious than the encapsulation concept and, therefore, more realistic.

The finally selected concept includes sealing of the major disposal areas and extensive backfilling of open cavities. It ensures the static stability of the mine, it minimises the remaining open volume accessible to possibly intruding brine from the cap rock and, therefore, also the formation of additional hollow space through dissolution processes. Furthermore it enhances the overall safety because the backfill material is an additional transport barrier with a high sorption capacity for radionuclides. The sealing of the major disposal areas guarantees well defined barriers between the waste and the potential locations of possible future brine inflow into the mine.

The safety and performance assessments for the selected closure concept were carried out by two teams with different models and computer codes [1,2]: Colenco used the code PROSA that had been developed especially for the concept of extensive backfilling of the Morsleben repository, while GRS applied the already approved code EMOS that had been used before in several other applications.

Features of the selected closure concept

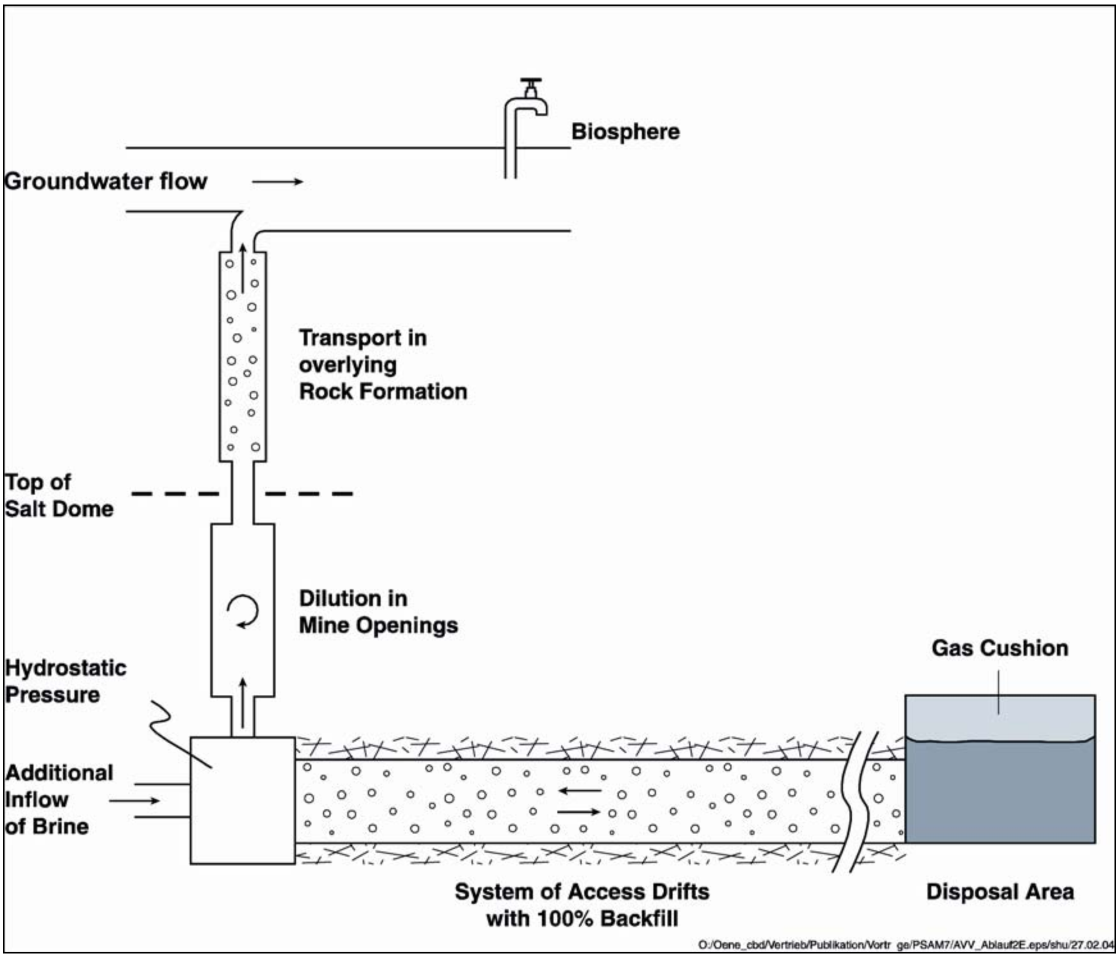
The main idea of the closure concept is that not a single engineered barrier or process prevents radionuclides from reaching the biosphere. Instead, the entire system exhibits a barrier effect through a partially redundant combination of several processes. These processes are as follows:

- The extensively backfilled repository has a high resistance to movement of brine. A presumed inflow of brine from the overlying rock formation into the mine and particularly into the waste caverns will be delayed and restricted in volume. Also, the structure of the mine will not change significantly as a result of salt dissolution.
- The disposal areas are – as far as possible – sealed off from the rest of the mine by specially designed seals of salt concrete.
- Sorption and limited solubility hinder part of the radionuclides from being released from the waste chambers.
- The extensive backfilling of the mine with a material of low compressibility structurally stabilises the mine, thereby reducing the convergence of the remaining cavities. As a result, the later discharge of previously intruded and potentially contaminated brine out of the confines of the repository will be low.
- The permeability of the concrete backfill is generally higher than that of the surrounding intact salt rock formation, but still is low in absolute terms. Therefore, brine movement and radionuclide transport will take place predominantly through the backfill and the adjacent zones of disintegrated salt rock with low flow rates. The backfill provides a medium for sorption at least for some of the radionuclides. However, no credit is taken from this process in the safety assessment.
- In addition to the still remaining open volumes, the porous backfill material itself allows for temporary storage of at least part of the gas generated in the repository, thus reducing the potential pressure build-up.
- While migrating through the backfilled caverns, drifts and shafts of the salt mine to the location where mine and rock overburden are in hydraulic contact, the radionuclide concentration will decrease due to dilution with non-contaminated brine.
- Outside of the salt formation, the transport of radionuclides through the cap rock and the overlying rock formations delays the release into the biosphere where mixing with the near surface groundwater further reduces the radionuclide concentration.

Safety assessment concept

The safety analyses are mainly based on the potential scenario of brine intrusion. This scenario is characterised by the following processes: Brine infiltrates through the cap rock, disintegrated zones of the salt rock barrier and the backfilled and open cavities of the mine. Brine which has interacted with potash salt will slowly corrode the seals made of salt concrete. This can lead to slow brine intrusion into the sealed waste disposal areas and cause dissolution of radionuclides from the waste. Gas generated as a result of waste degradation is stored partially inside mine cavities, leading to a pressure build-up that reduces brine inflow into the disposal area. In a later phase, convergence of the cavities in the salt and further gas generation displace contaminated brine from the disposal chambers and the sealed disposal areas to other parts of the mine, where it is diluted, and finally squeezed out of the mine into the cap rock. The brine then crosses the cap rock and the overlying rock formations and enters the aquifer with subsequent transfer to the biosphere (see Figure 1).

Figure 1. **Schematic representation of model concept**



To assess this scenario the mine with its technical barriers, the geosphere and the biosphere are represented in a single model which describes all important processes and covers a multitude of possible system configurations. Conceptual simplifications and a semi-analytical approach result in short computation times. Both codes, PROSA and EMOS, calculate the evolution with time of radiation exposure for a given set of parameters taking into account the processes outlined above.

The safety assessment concept was developed independently by each of the two groups. From time to time, intermediate reports were written and compared. Thus, both teams kept their independence, and a critical review of each others' results was possible.

For the final safety assessment the two teams used the same data set but different programmes, i.e., EMOS and PROSA. The models mainly differ in the conceptual simplification of the mine building, in the assumptions on the composition of brine which is in contact with the seals of the disposal areas, in the conceptualisation of the two-phase processes and of the water flow in the overburden and the aquifer. As well, there are some differences in the conceptualisation of the transport of $^{14}CH_4$ in solution and in the gas phase.

Parameter values and distribution functions

In spite of the simplifications of geometry and the processes involved,[1] more than 200 parameters are necessary to describe the system, not including the various specific parameters for the almost 50 radionuclides (inventories, half-lives, solubility, sorption, dose-coefficients). Some of these parameters are well defined. But most of the parameter values cannot be characterised exactly and are therefore considered with a rather large bandwidth. The uncertainty of some groups of parameters (e.g. sorption, convergence of cavities) which are internally correlated can be treated in a simplified manner by applying a common variable factor to the respective reference values. Additionally, different possible system developments, several potential migration pathways and patterns of fluid movement are mapped in the bandwidth of specific characteristic parameters.

The response of the system to parameter variations is highly non-linear. Different processes may either enhance or compensate each other so that the variation of a single parameter can change the result in either direction, depending on the other parameter values. Because of this, the common practice of selecting a "conservative" parameter set for the safety assessment is not possible.

In addition to deterministic calculations, a probabilistic approach for the safety assessment has been adopted. Each parameter is defined by a reference value, a maximum and a minimum value and by the type of its statistical distribution function (uniform, log-uniform, normal, log-normal or triangular). In order to analyse the influence of parameter uncertainties on the calculated radiation exposures, Monte Carlo approaches have been applied using independently the codes PROSA and EMOS. A large number of realisations were performed with simultaneous variation of all parameters. Thus, very pessimistic parameter combinations have been selected according to their probability, the latter being determined by the statistical distribution functions.

Some of the distribution functions are given directly by the known uncertainty of measured values (e.g. the geometric data related to the volume of cavities), some are the results of other probabilistic models,[2] and many had to be defined by expert judgement. Special scenarios are covered by extending the bandwidth of the distribution function of specific parameters. For example, the failure of a seal can be simulated by assigning an artificially high value to its initial permeability, resulting in an early chemical alteration and degradation because of faster access of (corroding) brine. A highly conservative definition of the corresponding distribution function inflates the probability of this scenario, yielding a relatively large number of "scores" with different combinations of the other parameters.

Results of the reference case

The calculated exfiltration of radionuclides within the first several thousand years is a consequence of the conservative assumption that the entire mine outside the sealed areas is instantaneously saturated by inflow of brine from the cap rock. Because there is some – relatively small – amount of radioactive waste in disposal areas that cannot be sealed against the rest of the mine and since no credit is taken from the delay of radionuclide transport outside of the sealed areas, the calculated exfiltration of non-adsorbing radionuclides into the biosphere begins after a delay of a few

1. Examples of simplifications: The "model disposal cavern" represents several interconnected disposal caverns and other cavities within the sealed area. All (sealed) connections between a disposal area and the rest of the mine are grouped together into a single "model seal".

2. For example, the parameters for the gas generation have been deduced from the results of an independent numerical model for gas generation. This model includes the uncertainties of inventories, reaction rates etc. and was also applied using a Monte Carlo approach.

hundred years only. This delay is caused by the transport through the cap rock and overburden. The main part of the disposed radionuclide inventory is, however, in the sealed disposal areas. The seals will withstand corrosion by brine for tens of thousands of years.

This can be seen in the model results (Figure 2). Both models show an early exfiltration with a maximum between several hundred and about one thousand years originating from the waste in the unsealed disposal areas and a second release starting after several ten thousand years. The latter stems from the waste in the sealed disposal areas after the shorter seals are corroded.

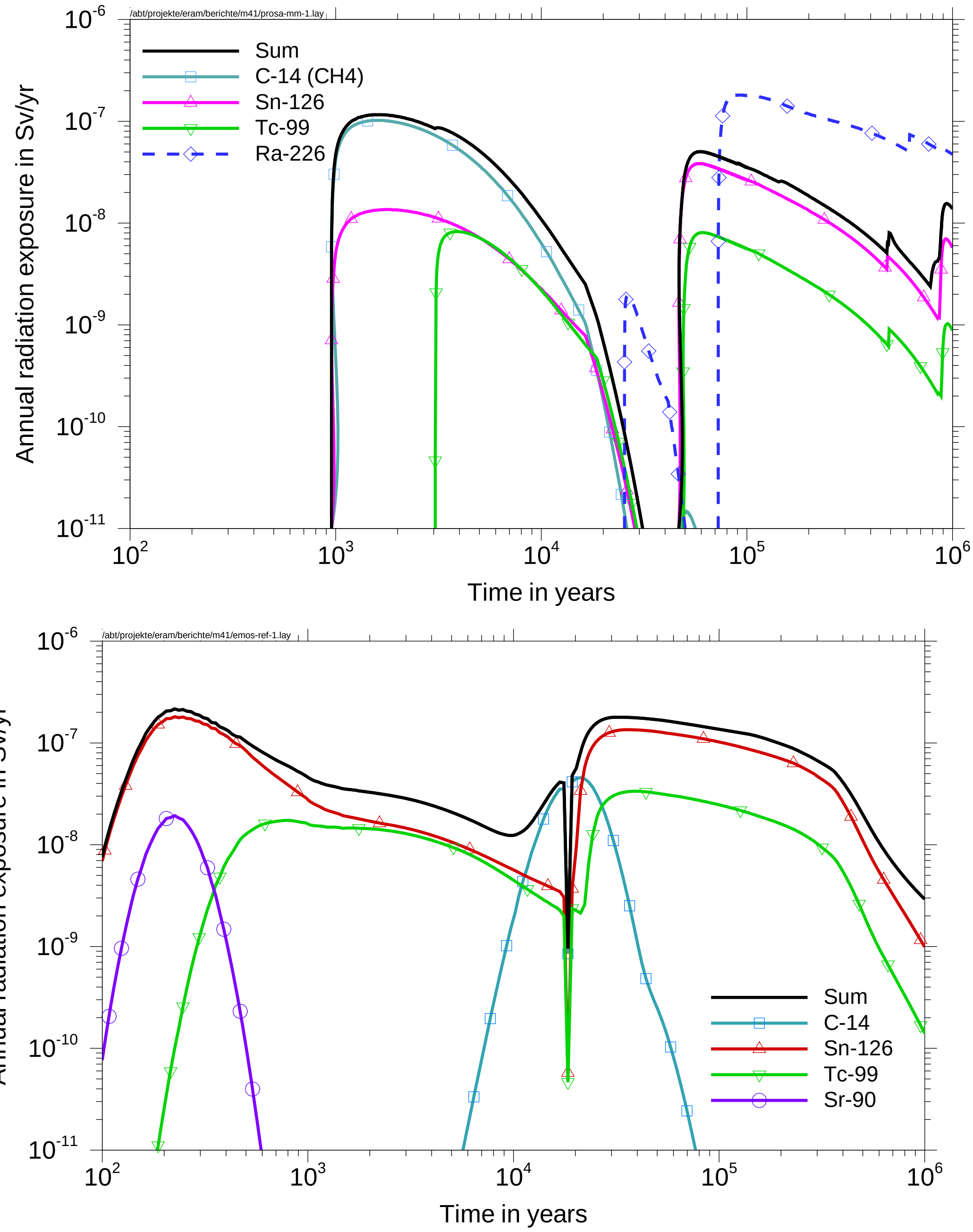
The differences between the results of EMOS and PROSA are caused by some differing conceptual assumptions. Consistent with the other model assumptions, in EMOS the gas that is generated from the corrosion of steel and microbial degradation of cellulose outside of the sealed disposal areas is considered for the displacement of contaminated brine from the mine, which results in a high flow rate of contaminated brine through cap rock and overburden and a rather short corresponding transport time during the first hundreds of years. Thus, even rather short-lived radionuclides like ^{90}Sr are released to the biosphere according to the model calculations with EMOS. Based on the more realistic assumption that gas which is generated in this early period cannot really displace contaminated brine into the cap rock the gas generation outside of the disposal areas is disregarded in the calculations with PROSA. This results in lower flow rates and longer transport times during the first few hundred years. After about 1 000 years, when the calculated release into the biosphere begins, short-lived radionuclides have decayed, so that no significant release of ^{90}Sr is calculated with PROSA.

Another difference is evident in the calculated times of the beginning release from the sealed disposal areas. According to EMOS, this release starts after less than 20,000 years, with PROSA this time is more than 40,000 years. This results from different assumptions with respect to the composition of the brine which is in contact with the seals. In the reference conceptualisation of EMOS it is assumed to be IP21 (“Q-Brine”), a product of saturation with carnallitite. In the reference conceptualisation of PROSA it is assumed that the brine has only 50% of the magnesium content of IP21 because the accessibility of carnallitite is limited as a consequence of the backfill measures and because some magnesium will be consumed by the reaction of the brine with backfill material on its way from the carnallitite to the seal. A lower magnesium content of the brine results in a lower corrosion capacity of the brine and a longer corrosion time for the seals of the disposal areas.

A further difference is caused by a conservative approximation within PROSA for the transport of radionuclide chains through cap rock and overburden. This results in a fictitious release of ^{226}Ra (dashed line in Figure 2), whereas with the transport times through cap rock and overburden resulting from retardation capacity and flow rate, the release of ^{226}Ra into the biosphere should be negligible as calculated with EMOS.

Besides these differences that result from the differing assumptions and conceptual simplifications, the model results of EMOS and PROSA agree well. This applies to the magnitude and the time of the maxima as well as to the relative contribution of the relevant nuclides (Figure 2). Several alternative scenarios for the transport of radionuclides in solution have been calculated with EMOS and with PROSA. Without discussing the details here, it can be concluded that the results of the two models correspond well. This enhances the confidence in the safety assessment.

Figure 2. **Radiation exposure in the reference case for the transport in solution; calculated with PROSA (above) and EMOS (below)**



Results of the probabilistic calculations

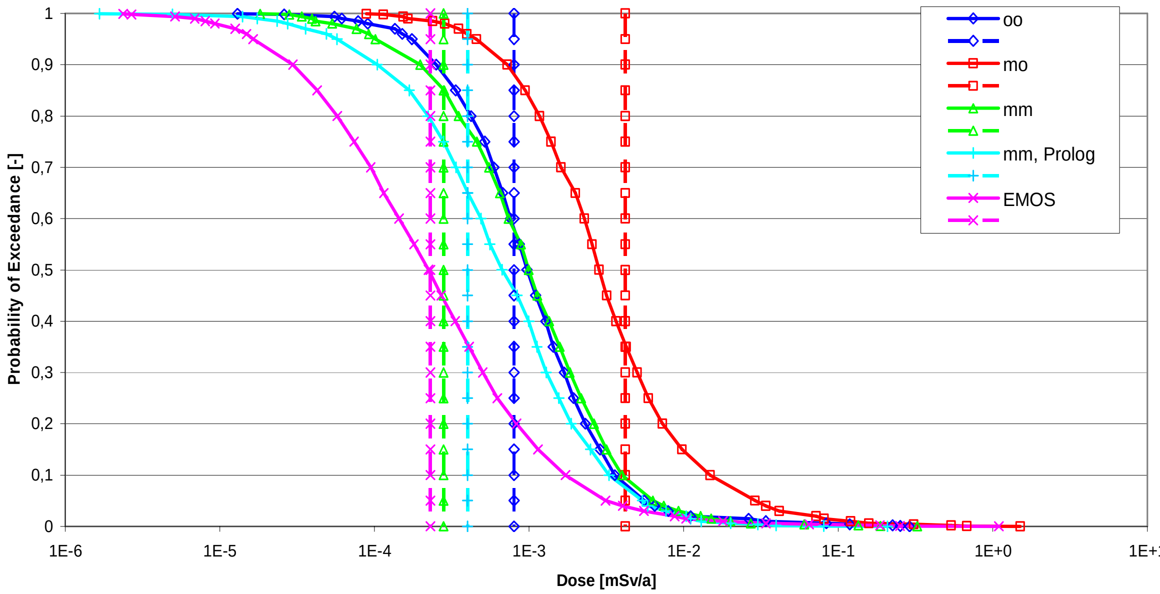
With EMOS, 2 000 randomly chosen parameter sets were calculated. With PROSA, four different conceptualisations were considered regarding the two-phase behaviour of the gas in the disposal areas and the time when the mine structures outside of the sealed disposal areas are filled with brine from the cap rock. For each of these conceptualisations 1 000 randomly selected parameter sets were calculated. The results are shown in Figure 3.

The four different conceptualisations of PROSA are denoted as follows:

"oo"	gas generation does not influence the inflow of brine into the disposal areas and does not enhance the displacement of brine from the disposal areas
"mo"	gas generation enhances pressure build-up in the disposal areas; the gas hinders first the inflow of brine and later expels brine from the disposal areas; the gas cannot escape from the disposal areas
"mm"	gas generation enhances pressure build-up in the disposal areas; the gas hinders first the inflow of brine into the disposal areas; gas and brine are later simultaneously expelled through the seals
"mm Prolog"	like "mm", but delayed inflow of brine from the cap rock into the mine

More than 99% of the realisations are well below the dose constraint of 0.3 mSv/a, with the few remaining realisations being below 1 mSv/a. Detailed analysis shows that the few realisations which slightly exceed the dose constraint of 0.3 mSv/a result from a combination of parameter values lying close to the extremes of their bandwidths. A less conservative choice of the parameter distributions or of model assumptions would clearly avoid even these few calculated exceedings of the constraint.

Figure 3. **Probability of exceedance for a given dose value; result of the probabilistic calculations with EMOS and with PROSA. Dashed line: maximum dose of the corresponding reference case**



The different possibilities for the behaviour of the gas are represented in EMOS by a two-phase parameter, the air entry pressure for the gas, which controls the release of gas from the disposal areas through the topmost seal. Thus, the probabilistic calculations with EMOS match to a certain degree the calculations with PROSA for the three different conceptualisations of the behaviour of the gas. Because of that, the realisations with EMOS cover a broader range of system behaviour than one set of realisations with PROSA, and the slope is not as steep. With EMOS, the calculated doses are generally lower than with PROSA. This is partly due to the conservative approximation within PROSA for the transport of ^{226}Ra through cap rock and overburden (cf. Figure 2 and the discussion in chapter "Results of the reference case").

Especially for the probabilistic calculations with PROSA, the median dose values (probability of exceedance = 0.5) and the dose value calculated deterministically for the reference parameter set can differ significantly. This is due to the high non-linearity of the system and the asymmetric distribution functions for several parameters.

Radionuclide transport in the gas phase

Regarding the radionuclide inventory in the Morsleben repository and its chemical form, only the transport of $^{14}CH_4$ in the gas phase from the waste, through the mine, cap rock and overburden into the biosphere has to be considered. The two project teams of Colenco and GRS developed different conceptualisations for the transport of radionuclides in the gas phase. Colenco considered the concentration of $^{14}CH_4$ in the gas phase in the mine, the pressure build-up due to gas generation and convergence and 2-phase processes in the cap rock (the latter were modelled with TOUGH2). GRS only considered the volumes and the gas generation rates in the mine and disregarded any barrier effect of the cap rock. GRS calculated maximum release rates of about 10^9 Bq/a. Colenco calculated 10^6 Bq/a in the reference case and $3 \cdot 10^7$ Bq/a in the worst case; the sensitivity to the 2-phase parameters was low.

Colenco and GRS used the same biosphere model for the reference scenario with the oxidation of $^{14}CH_4$ to $^{14}CO_2$ in the unsaturated zone above the water table and the release of the latter into the air above an agricultural area. On top of this, both teams developed their own alternative worst-case biosphere model. In all cases the calculated doses were well below the constraint of 0.3 mSv/a.

Conclusions

Safety assessments for the Morsleben radioactive waste repository (ERAM) have been performed independently by two groups using different conceptual models and codes but, as far as possible, identical parameters. Calculated radiation exposures are well below applicable dose constraints. The results of the two models correspond satisfactorily and the results from probabilistic calculations corroborate the ones from the deterministic calculations. Differences in the results from the two independent groups are well explainable and related to the different model conceptualisations. These findings significantly enhance the confidence in the modelling results and contribute significantly to the safety case for the geological disposal of radioactive waste at the Morsleben site.

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EUROPEAN PILOT STUDY ON THE REGULATORY REVIEW OF THE SAFETY CASE FOR GEOLOGICAL DISPOSAL OF RADIOACTIVE WASTE

F. Besnus
IRSN, Institut de radioprotection et de Sûreté Nucléaire, France

J. Vigfusson
HSK, Swiss Federal Nuclear Safety Inspectorate, Switzerland

R. Smith
Radioactive Substances Regulation Environment Agency, United Kingdom

V. Nys
AVN, Association Vinçotte Nucléaire, Belgium

G. Bruno
EC, European Commission – Directorate General for Energy and Transport

P. Metcalf
IAEA, International Atomic Energy Agency

C. Ruiz-Lopez
CSN, Consejo de Seguridad Nuclear, Spain

E. Ruokola
STUK, Radiation and Nuclear Safety Authority, Finland

M. Jensen
SSI, Swedish Radiation Protection Authority, Sweden

K. Röhlig
GRS, Gesellschaft für Anlagen-und Reaktorsicherheit mbH, Germany

P. Bodenez
ASN, Nuclear Safety Authority, France

Abstract

A number of countries within Europe are developing or giving consideration to the development of geological disposal facilities for the disposal of high level radioactive waste. The safety authorities in these countries are interested in exploring the possibility of a harmonised approach to the demonstration of safety of such facilities and the regulatory review of documentation providing such demonstration. As such and with technical support organisations and international bodies they have initiated a pilot study on how these elements should be presented in a safety case, for, *inter alia*, regulatory review and approval. It is envisaged that such a study will cover assessment of the site and

engineering, impact assessment and assessment of the management systems ensuring quality and will be compatible with internationally agreed safety standards, guidance and recommendations. It is foreseen that the safety case will evolve and mature as the project develops and this aspect has been considered within the pilot study, together with regulatory review and decision making at discrete milestones associated with the disposal facility development. The study is focused on regulatory expectations for the different milestones and addresses uncertainty management. The paper present the work carried out to date and the views for future work.

A summary of the findings of the European Pilot Study

A number of countries are developing or giving consideration to the development of geological disposal facilities for the disposal of high level radioactive waste and harmonising approaches to achieve a high level of safety for such facility becomes a foreseeable and necessary objective. To this respect, important steps have already been successfully taken by international organisations such as IAEA and OECD/NEA in developing internationally agreed standards, guidance, recommendations and collective opinions. Within the European Union, a Working Party on Nuclear Safety (WPNS) is currently analysing to what extent common approaches of waste management are implemented by EU Member States while the Western European Nuclear Regulators' association (WENRA) has started harmonising views concerning best safety practices for the predisposal management of radioactive waste. In parallel, France and Belgium have cooperated in developing ideas on the safety approach to geological disposal and reached common positions that were presented to an enlarged group of European regulators and International organisations. It was considered that valuable momentum had been created by the French-Belgium initiative and that efforts to develop common views should be enlarged to other interested countries within the European region. It was consequently decided to launch a pilot study, aiming at sharing experience and opinions on the expectations of the regulator for different elements of a safety case for geological disposal of radioactive waste at the different steps in a project to develop such a disposal facility. The pilot study is carried out by a group of regulators and technical support organisations from Belgium (FANC, AVN), UK (EA), France (ASN, IRSN), Finland (STUK), Germany (GRS), Spain (CSN), Sweden (SSI), Switzerland (HSK), as well as representatives of international organisations (IAEA, EC).

Although regulatory frameworks differ considerably between countries, the Working Group recognised that regulatory practice differs to much less an extent.

It is now widely accepted that development of a disposal facility and its safety case should take place in a step-by-step manner with well-defined decision points. The degree to which a step-by-step process is implemented in regulations varies from country to country, and so do the requirements concerning the involvement of regulators and licensing authorities at various decision points. The Working Group considers, however, that it is important to keep regulatory and licensing authorities and their technical support organisations informed about the state of development at each step and to involve them in the major decisions (e.g. about the disposal facility concept or about R&D priorities), no matter whether or not there is a formal requirement for doing so.

From a regulatory perspective, the key stages are:

- conceptualisation;
- siting;
- design;
- excavation/construction;
- operation;
- closure.

For the purposes of the pilot study, the group chose to focus on:

- the conceptualisation stage, during which a safety strategy must be developed together with a review of potential sites and design options as well as preliminary assessments in order to enable decision making on committing resources to the next stage of the project,
- the siting stage, for which the safety strategy must be confirmed as well as design suitability for given site(s) that have undergone due characterisation in order to make decision of site selection for hosting a disposal facility,
- the design stage that comprises design validation and safety assessment in order to license construction of the disposal facility.

The group considered that at each development stage each of the following aspects should be considered:

- facility design and the safety strategy;
- demonstration of site and engineering suitability;
- impact assessment;
- adequacy of management systems.

In this respect, the safety case presenting the arguments and supporting information and assessment related to the above aspects will have to be comprised of clear information, from the very beginning of a disposal project, covering the design options and the key elements upon which safety relies, together with a description of the preferred strategy to acquire progressively enough knowledge of the factors governing the containment and isolation capacity of the disposal system. The safety case will also need to accommodate uncertainties, as discussed further below. Given that a preliminary design and safety strategy is a necessary input for project implementation, the assessment of the soundness of the proposed options is essential to enable the project to move forward to the next step. The safety case must therefore be comprised of information, assessment and arguments, covering the three components mentioned above and aiming at identifying advantages and disadvantages of the chosen design and strategy in terms of safety throughout the project development. The elements to be described are :

- **that related to assessment and demonstration of the site and engineering suitability** ; this comprises all information concerning site and design selection, characterisation and validation with regard to safety. The site and engineering assessment must include a description of the functions assigned to each component of the disposal system (for both the operational and post-closure phases) and an analysis, through a performance assessment, of the capacity of these components (including the host rock) to fulfil their given role. In this respect, the situations and phenomena that may affect system performance, both internal and external (heat, corrosion, radiolysis, mechanical stress, criticality, seismicity, climate change, etc.) must be identified and quantified so as to assess the system behaviour and robustness. The site and engineering assessment also address elements of construction feasibility and reliability. In all these respects, the design and site performance will need to be justified and the uncertainties remaining at the particular stage of the project will be identified;
- **that related to impact assessment** in terms of radiation dose, risk, some combination of both or other entities indicating potential impact; this comprises appropriate modelling and data selection so as to assess exposures that might arise from facility operations and long term evolution with a sufficient level of confidence. It requires clear substantiation that assessment of selected scenarios provides a conservative estimate of the impact of the facility. It also requires sensitivity analysis so as to identify key dependencies on parameter values and assumptions together with evaluating the effect of uncertainties;

- **that relating to demonstrating the adequacy of management systems** associated with the project to assure an adequate level of quality in respect of all safety related aspects of the project; this comprises in particular organisational arrangements for implementing the project, planning of major actions to be taken during the different steps, elaboration of operational and control procedures, record keeping and review.

All these elements are of course inter-related and form together the necessary bases for demonstrating the safety of a disposal facility and providing input to decision making.

The group considers that this structure should be maintained through every stage of the step-by-step process, with the content of the safety case being progressively developed as the project proceeds. For each key step of decision making, a decision should be taken only if structured information on all important elements of the disposal system is available and the supporting safety assessment is performed, even if at preliminary steps only partial information is available. For example, selecting a site cannot be based only on consideration of geological data but also requires an analysis of site and design compatibility, an assessment of how the whole system works together to achieve safety and consideration of whether appropriate techniques are available to realise the system.

Uncertainties concerning the safety of disposal facilities are unavoidable due to the complexity of the phenomena of concern and the scales in time and space under consideration. Their management is central when developing a disposal system and assessing its safety. For this reason, the issue of uncertainties and their management has been chosen for a more detailed examination in the frame of the Pilot study in order to identify the level of commonality on this subject among the participating countries, to better understand differences, and to propose some common grounds for guidance. A self-contained document has been developed and appended to the Pilot study final report, focusing on the handling of uncertainties in the context of the overall safety strategy, safety assessment and evaluation of compliance with safety requirements. The document elaborates on the following concepts:

- **Safety Strategy.** Conscious accounting for uncertainties and analysis of their possible consequences is a required part of any safety assessment for a radioactive waste disposal facility. Within a step-by-step approach to disposal facility development, this includes providing a register of significant uncertainties and a management process for assessing and, where appropriate, avoiding, mitigating or reducing them**.**
- **Assessment strategy.** Regarding the effect of uncertainties on the safety assessment, emphasis is placed on those elements of the assessment which are not yet fully included within guidance provided by international organisations such as IAEA or NEA, namely**:**
  - the strategy of scenarios;
  - the role of deterministic and probabilistic approaches; and
  - the role of best estimate, conservative and pessimistic models and parameter values.
- **Evaluation of compliance with safety requirements in the presence of uncertainty.** Although the approach for compliance evaluation differs considerably depending on the country, the issue remains that many uncertainties in the post-closure safety case cannot reliably be quantified. Calculated doses or risks can only be regarded as broadly conservative indicators rather than anything more definite or concrete and, accordingly, the post-closure safety case needs to be based on multiple lines of reasoning**.**

In conclusion, the Pilot study offers examples of the ability to find common ground among the regulators from different European countries, both in terms of the general framework and for particular topics, e.g. uncertainty management. This suggests the possibility of work to find further areas of common ground and, perhaps, areas for harmonisation.

FROM INITIAL SAFETY CONSIDERATIONS TO A FINAL SAFETY CASE: FORMS AND PURPOSES OF SAFETY ANALYSIS DURING A DISPOSAL PROJECT'S LIFETIME

J. Vigfusson
HSK, Switzerland

Abstract

During preparation and realisation of a radioactive waste disposal project a multitude of decisions are taken which require input from an assessment of the radiological safety of the disposal facility. Depending on the stage of the project and the purpose of the assessment, the safety assessments that are needed – or are possible – vary in comprehensiveness and in the choice of safety indicators. The paper compares the suite of safety considerations employed from the beginning of the search for a suitable radioactive waste repository site until the final closure of the repository. The reference material mostly derives from the regulatory basis and actual planning of the Swiss repository projects but illustrative examples are also taken from other national programmes.

Introduction

The planning and realisation of a geological repository for radioactive waste is a long-term undertaking that usually requires extensive involvement of political decision makers. It has become common understanding that such projects have to proceed in a stepwise manner, where the decision to move ahead to begin the next step is supported by a careful review and evaluation of the project with respect to experience gained in the previous step and need for adjustments of the planned continuation. At these halting points, there is the opportunity to take advantage of the evolution of the scientific and technical basis as well as the clarification of relevant societal questions in order to improve the final outcome.

The arguments showing the long-term safety of the repository are important parts of each such evaluation. As time goes by, there will be a succession of safety cases prepared for the repository project, starting perhaps with comparatively simple assessments to support candidate site evaluations and eventually progressing to a sophisticated and firmly founded safety case supporting the decision to close the facility. Whether early or late in the process, on a level commensurate with the actual step, each safety case will address the post-closure phase to provide "a synthesis of evidence, analyses and arguments that quantify and substantiate a claim that the repository will be safe after closure and beyond the time when active control of the facility can be relied on." [1]

In the following, I shall focus on the evolving role and form of the safety case from the beginning of the search for a suitable repository site on, during the licensing, construction and operation phases, until the decision is taken to close the repository for good. The background to this discussion is the actual situation of the Swiss disposal programme, which is now entering a new stage. After the successful feasibility demonstration of the disposal of spent fuel and high level waste, submitted by Nagra in December 2002 [2], which received a positive review by the regulatory body in August 2005 leading to its acceptance by the Swiss government in June 2006, the focus of the programme is shifting from the demonstration of feasibility to the implementation of one or more repositories. The next several years will be dedicated to the selection of suitable sites for the repositories to be built. With the site selection process thus coming into focus, the role of safety considerations during site selection emerges as an interesting subject. In contrast to the mature safety cases that will be presented from the beginning of the licensing process on, the site selection process will have less elaborate safety cases growing with it, evolving from simple assessments based on preliminary or generic data to more extensive site specific assessments as the site candidates are carried forward and approach a possible selection. Given this background, the discussion will mainly be based on the Swiss regulatory system and repository project studies, although some reference will be made to other countries. The discussion will not be exhaustive in the sense that it would cover all situations in which a safety case has been needed in the past, one notable omission being the already mentioned feasibility demonstration in Switzerland. Rather, the choice of topics will be influenced by the more recent involvement of the regulator in shaping the Swiss site selection procedure.

Stepwise development of repositories

The licensing system for radioactive waste repositories in Switzerland implements a stepwise development, where each step requires a separate license. After the general license, which fixes certain outlines of the repository project, such as the site, the waste categories and the maximum inventory, a construction license, an operating license and a license (or rather a governmental order) to go ahead with the closure are needed. For each of these licenses, the application must be accompanied by a new or updated safety case. But also well ahead of the general license, safety cases are needed that are reviewed by the regulator. Switzerland is in the course of developing a site selection procedure that should lead to the selection of a good repository site in a transparent way with early involvement of the public. The procedure description not only specifies the steps to be taken and the consultations to be done during the process but also lays down and explains the criteria that govern decisions at each step. While there are not only safety considerations but also land use planning and socio-economic considerations to be recognised, the procedure ensures that the safety criteria always take priority. Other criteria are only allowed to decide between site candidates that are considered to offer similar levels of contribution to the safety of the repository. Two things should now have emerged: Safety cases will repeatedly be needed during the selection process, and there has to be an agreed method for deciding in a transparent manner when two sites are considered to offer similar contributions to safety. The safety cases developed during site selection and during the lifetime of the repository evolve in their precision and sophistication as the project unfolds. We shall return to that evolution in the following section.

Whether or not the licensing system requires it, many countries are experiencing a succession of safety cases of increasing sophistication, as preparations proceed towards repository operation and later closure.

In Sweden, the implementer is selecting a site for a spent fuel repository. Early in the site selection process a generic safety case was assembled (SR 97) [3], based on data from three different sites. The purpose of this safety case was to demonstrate the feasibility of finding a site in Swedish bedrock, demonstrate the methodology for safety assessment and give feedback to guide site investigations and engineered barrier development. Later, the site selection was guided by preliminary safety evaluations that compared the site characteristics with predefined criteria. For the most likely candidate sites a preliminary long-term safety case was developed in 2006 (Project SR-Can) [4]. The initial purpose of this safety case was to support a license application for an encapsulation plant for spent fuel. Later the emphasis shifted to providing feed-back to the repository design, guiding further site characterisation and generally serving as preparation for the safety case required for the next step. Thus, based on the results from additional site characterisation, this safety case will soon be followed by a refined safety case (SR-Site) supporting the application for a repository siting and construction license.

In Finland, the first licensing step for a repository for spent fuel in Olkiluoto, the Decision in Principle, was completed in 2001. The licence application was based on the safety case TILA-99 [5] covering the four candidate sites involved in the last step of the site selection process. Earlier, there had been preliminary safety assessments, the most important being TVO-92 (1992) and TILA-96 (1996).

In the US, the Yucca Mountain project will go through a succession of license steps, including the construction and the operating licences and a license amendment authorising the closure. At each step a safety case will be needed. Ahead of licensing, a total system performance assessment was already needed in order to support the 2002 recommendation of the Secretary of Energy to the President that the Yucca Mountain site be developed as the site of a repository for spent fuel and high level waste.

In Germany, a site selection procedure has been proposed by the advisory group AkEnd [6]. The procedure definition in particular mentions two steps involving each a safety case: an orienting safety evaluation based on site characterisation from the surface and a safety case based also on the results of an underground survey. For the selected site, a detailed safety case is also needed as input to the (single step) licensing process.

Role and contents of the safety case during site selection

The site selection process in Switzerland

The site selection procedure [7] being developed in Switzerland defines three stages1. During the first stage, a number of candidate site regions are identified by a systematic process starting, in principle, from an “empty map” of the country, but making appropriate use of already established geoscientific data. The candidate site regions are selected on the basis of an evaluation of their likely contributions to the safety of a repository. In the second stage a comparison is made of candidate sites from the viewpoint of their contribution to the safety of a repository and a ranking based on criteria relating to land use planning and socio-economic aspects. This ranking is between candidate sites with comparable safety contributions. As a result, at least two candidates are identified that enter the third stage. In the third, final stage, a more thorough assessment and comparison of the candidate sites is made. A site is then selected, for which an application for a general license is submitted.

During this process, there are three occasions on which a safety case is required. The earliest safety case, the generic safety evaluation, serves a role right at the beginning of the first stage of the site selection process. This may at first be surprising, since at that time particular sites have not been identified and there is a lot of freedom in how to design the repository. But a look at the details of the early steps in the site selection process will make the purpose of the safety evaluation clearer. This point will be taken up in the following sub-section. In the second stage of the selection process, the candidate sites are compared and their number reduced. This happens in two steps: First, it is clarified whether the different candidate sites hold comparable promise for a safe repository. Then, within the safest group of comparable sites, these are ranked on the basis of criteria relating to land use planning, regional economics and infrastructure. To inform a decision on the comparability of different sites with respect to their contribution to the safety of a repository, site relevant safety assessments are

1. The site selection procedure was still in development at the time of the writing of this paper. The description refers to the draft process description as of November 2006.

needed. This is part of the preliminary safety case for each site. The final choice of the repository site in stage three again requires a thorough and detailed comparison of the two or more candidate sites that remain in this last stage of the selection process. This will require a more detailed and complete safety case for each these candidate sites. It is the intention that, for the selected site, the safety case developed in the third stage should not need any major updating in order to serve in support of the application for the general license. The three safety cases mentioned clearly have different functions and will build on data bases of different sophistication. Each will be discussed in the following subsections.

Generic safety evaluation

The site selection procedure takes a geological overview over the whole country as a starting point for a stepwise process of narrowing in on the finally identified possible candidate sites. The process of narrowing in is based on a set of safety relevant geological criteria that have been publicly discussed and agreed in advance. For each site or region, the criteria are evaluated on a scale of qualitative statements, such as “very good”, “good”, “adequate”, “less adequate”. There are several reasons for using a qualitative scale in the early stages of the selection process. One reason is that the geological data may lack the precision to make a detailed quantitative ranking meaningful. Another reason is that the safety of a repository on the site will depend on the set of geological criteria as an intertwined whole, so that a comparison of one or two parameters in isolation is not a good basis for comparing safety at different sites. Also, in the very first steps, promising geological areas are identified primarily on the grounds of the large scale tectonic situation. Here only a subset of the criteria is applicable, and a precise ranking would not be fully relevant for the comparison at the level of sites. But why is then a safety case needed?

For a site selection process to have public acceptance, it has to be clear in advance and auditable afterwards, how the experts arrive at their qualitative judgement. It must be agreed how much of a certain property corresponds to the evaluation “good” with respect to that property. In analogy with the description of the assessment basis in a safety case, in the very first step of the site selection process the implementer must declare what kind of repository he will be looking for: what is the inventory that should be accommodated, the safety concept, the intended design of the engineered barriers. Based on his choices, the implementer is required to do a “generic safety evaluation” that illustrates the necessary isolation time of the waste and shows the contributions to confinement and isolation that is to be expected of the engineered barriers. From this information, the implementer must draw quantitative conclusions on the levels of safety relevant qualities of the host and regional geology that correspond to each statement on the qualitative scale. This may take the form of a single parameter bandwidth, or there may be a more involved evaluation that takes account of the inter-dependence of the different geological properties in their contribution to safety. What is important is that the qualitative judgements of the site properties will thus depend on the assumptions made regarding the waste inventory and the properties of the engineered barriers.

An expert judgement is often needed, and so also in assembling the qualitative judgements of the single parameters or properties into a final qualitative judgement of the candidate site. The implementer will have to explain why he considers some properties of the host rock to be more important than others when he derives his final judgement. The quantitative characterisation and gauging of the qualitative scale based on the generic safety evaluation however helps to make this process more transparent and auditable. Sites or regions that score poorly in the final judgement will not qualify for the following stage.

The generic safety evaluation obviously cannot be site specific. It takes as input the information about the repository concept including the intended inventory and the near field barriers. Combining this with models of a likely geological barrier configurations and hydrogeological situations, the calculations of nuclide release and dose consequences are used to inform and guide decisions on how quantified properties of a candidate site should be evaluated on the qualitative scale. Comparing with generic information about the properties of the barrier materials, the calculations may also lead to adjustments of the inventory or near field barrier concept. It will not be required that this safety evaluation has all the components typical of a fully developed safety case. It will however at least contain a base case analysis resulting in a calculated dose that is compared to the regulatory constraint and it will need to take due account of the effects of variations in the parameter values. Compared to the usual safety cases this safety evaluation is aimed at solving the inverse problem: given that the safety criteria are to be satisfied, to derive the minimal requirements to be made regarding the properties of the site.

Preliminary safety case

The preliminary safety cases are developed during the second stage of the site selection process. They serve to support decisions on whether to consider the contributions to safety offered by different sites as comparable. The further site selection steps are then restricted to choosing between sites that belong to the safest group of comparable sites.

While most safety cases are well suited for demonstrating that a certain repository system offers sufficient protection to comply with internal or regulatory criteria, it requires more care to use them to compare different repositories. Depending on the type of the analysis, e.g., probabilistic or deterministic, it may be more or less easy to extract information on the expectable level of safety and on the possible consequences of the uncertainties about material properties and processes.

Performance calculations for supporting a licensing application usually must show that a certain maximum allowed measure of consequences is not exceeded. Uncertainty in parameters and models can then be handled by assuming pessimistic values for the parameters and model properties, and thus calculating at an upper bound for the consequences. An assessment containing many pessimistic assumptions may, however, not give a good indication of the most likely performance of the repository. For the sake of comparison of sites, this may not be a problem if the sites are similar and the analyses rely on the same or comparable pessimistic assumptions. But, what if the sites are very different as regards the host rocks and the regional geological settings? The question then comes up whether it would be a better choice to base the comparison on a best estimate of the repository performance, avoiding pessimistic assumptions to the extent possible.

Comparing best realistic estimates of the performance of the repository, based on a realistically expected system evolution, will in theory lead to a realistic assessment of the expectable difference in performance of the sites. But this does not suffice in order to decide between the sites. Safety analysis is as much an analysis of uncertainties as it is the analysis of processes and their outcome. A probably safe site that under less probable circumstances could fail with dire consequences may be less preferable than a probably less, but sufficiently safe site that does not have failure modes that lead to grave consequences. Thus robustness of the protective capability of the site is an important property to be taken into account when comparing sites.

Thus it becomes clear that the preliminary safety case will need to be based on a performance assessment that:

- takes account of the site properties, thus differentiating between sites;
- either exhibits the expected performance of the repository system, or is accompanied by a thorough discussion of the pessimistic assumptions made;

- explores and quantifies the robustness of the safety features of the site by an analysis of uncertainties, most likely involving different scenarios.

The preliminary safety case as described for the Swiss site selection process [7] in addition contains an assessment of the amount of further research and site exploration that would be needed before an application for a general license could be prepared.

The preliminary safety case has most of the standard components of a fully developed safety case, although there will perhaps be more emphasis than usual on the expected performance, and very rare features and events will hardly need to be discussed in great detail. Even so, the comparison of different sites will not be easy and will involve a good measure of judgement. Nevertheless, the advantage of the safety cases is the more thorough evaluation to base that judgement on. This evaluation also helps to clearly communicate the judgement to others.

Safety Case for the general license application

The decision to go ahead and enter the licensing process with a selected site is the final and most important milestone in the site selection process. This decision must be well founded, having followed from a scientifically and politically sound selection process and having taken advantage of all available information to support a thorough analysis of the final candidate sites. The safety cases will need to show a good degree of comprehensiveness as regards the site data collected and the quantitative discussion of the remaining uncertainties. Ideally, for the selected site only relatively minor further developments should be necessary in order to use the safety case to support the general license application. As will be discussed later on, stepwise licensing allows that some safety relevant questions remain open at this stage and, in particular, the Swiss regulator assumes that the safety case for the general license is mainly based on site data coming from site investigations from the surface.

Evolution of the safety case during the lifetime of the repository

When is a safety case required?

An answer to this question is given by the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management [8]. According to Article 13 of the Convention, it is necessary at the time of siting to evaluate all relevant site-related factors likely to affect the safety of the facility during its operation and after its closure and to evaluate the likely safety impact of the facility on individuals, society and the environment, taking into account possible evolution of the site conditions after closure. According to Article 15, systematic safety assessments covering both the operational and the post-closure phases are required before construction of the facility. The safety assessment for the operational phase must be updated and further detailed, if deemed necessary, before the operation of the facility.

The IAEA Safety Standards document: Geologic Disposal of Radioactive Waste, Safety Requirements [9], contains a more extensive discussion where it is specified that a safety case and supporting safety assessment must be prepared at each step in the development, operation and closure of the disposal facility. In this IAEA document, steps refer mainly to steps imposed by the regulatory and political decision making processes. The necessity of a stepwise development is of course not exclusively dependent on the question whether licensing is done in one step or several. The main advantage gained from the iterative safety evaluation is that the accumulation of knowledge as time goes on and perhaps also the change in attitudes and values in society can be taken into account as appropriate to improve the measures taken for ensuring long term passive safety of the repository.

Safety cases in stepwise licensing

The Swiss nuclear legislation requires a safety case to be presented at each licensing step. In principle, all these safety cases are “full-scale” safety cases, addressing all aspects expected of a safety case. Still the requirements on their completeness will be evolving. As already mentioned, the intention is to be able to base the licensing decision on best knowledge that is current at that time. More generally, the iterative production of safety cases must also be seen under the aspect of the management of uncertainties. It is permissible that non-trivial uncertainties are unresolved in an early safety case, provided that the possible impact of the uncertain features can be characterised and are found not to threaten the demonstration of safety. The safety case must contain a discussion of all relevant uncertainties. A safety case early in the licensing suite must also contain a discussion of the open questions remaining at that time and must draw up a roadmap of how these open questions will be addressed in the near future. The regulator, on the other hand, will in his review formulate his expectations as regards the closing of the open questions and may suggest corresponding licensing conditions to the licensing authority. The number and relevance of open questions regarding uncertainties should be insignificant in the later licensing stages of operational and closure licenses. A transparent analysis and management of uncertainties is beneficial both for the public involvement in the project and for the reviewability of the safety case.

Only one other example will be mentioned here. In Finland a Preliminary Safety Assessment Report (PSAR) must be presented when applying for a construction licence for a repository and a Final Safety Assessment Report (FSAR) for the operation license. In contrast to the safety case needed for the Decision in Principle for the spent fuel repository in Olkiluoto, which was based on the results of the site characterisation from the surface, the PSAR will be able to use results from the underground research facility ONKALO being built at the site. The FSAR will additionally profit from observations during the construction and will be able to refer to the design as built.

Conclusions

The development, operation and closure of a repository is a process which spreads over many tens of years. Safety analyses are needed early in the process, in order to clarify aims and guide site selection. Safety cases are repeatedly needed at major decision points during the project lifetime. The focuses of the safety cases change according to the purpose served. During the project history the refinement of the safety cases will also increase as the information gradually becomes more complete and precise and thanks to the development of better analysis tools. But there will be other changes too: the audiences will change, the general scientific background will evolve and so will the philosophical attitudes in society. The problem of dealing with long-lived radioactive waste forces us to think seriously about exotic concepts such as intergenerational equity across millennia, responsibility towards non-human biota and the importance of preserving a sound environment as a basis for sustainable life on earth. The stepwise development of the disposal facilities with the possibility of independent reviews and public involvement at each step will allow us to adjust and improve the projects as these developments unfold.

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DATA GATHERING AND LONG-TERM MAINTENANCE OF CONFIDENCE

Z. Nagy
Public Agency for Radioactive Waste Management, Hungary

Abstract

According to the Hungarian legislation safety assessment is required in the preparation-, establishment-, operational-, closure-licensing phase of repository development and during the operation of the repository the safety assessment should be reviewed regularly. It means that the safety assessment will be made many times during the repository developing process, between two consecutive safety assessments more decades may elapse and between the first and the "last" safety assessment more than one century will elapse. The Assessment of long-term safety is based partly upon geoscientific information. How can we maintain the confidence in the now gathered geoscientific information for many decades? What kind of metadata do we need to integrate into the multi-disciplinary geoscientific databases additionally for that purpose? The presentation sets forth the Hungarian practice and the up to now recognised existing difficulties.

Introduction

The life time of repositories is accompanied by safety cases. A safety case has to be made in the course of site selection and characterisation-, establishment-, operational- and closure phase, lastly finishing the active regulatory inspection. In compliance with this, between carrying out the first and last safety case many decades ore more than one century may elapse. We may guess that

- the later a safety case is carried out the more it will use from the consequences of safety cases carried out earlier;
- the deferred safety case will be based on previous safety cases;
- the level of confidence of the new safety case will be not lower than the confidence-level of the safety cases made earlier;
- the stage of uncertainties related to the new safety case will be lower than in the previous cases.

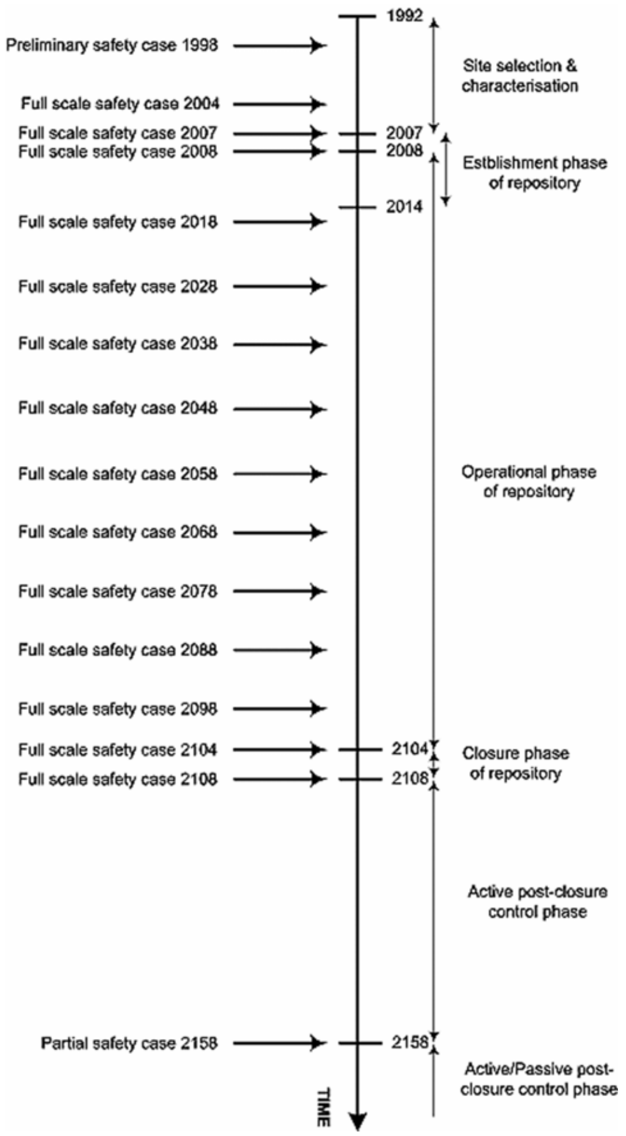
How are these expectations established? Will the data originated from many decades or 100 years old assessments really be used as the basis of the new safety case? Does the confidence of many decades old "results" remain on the same level? If not, what are the causes and consequences of this confidence-level decrease, what can we do to prevent or to postpone this confidence-level decrease? Are each part of the assessment the basis [1] for the same extent touched by decrease of confidence?

Background Law

According to the Hungarian legislation safety assessment is required in the preparation, establishment, operation, modification, termination, closure, and for the change-over to active or passive regulatory inspection licensing phase of repository [2]. During the operation of the repository the safety assessment should be reviewed regularly, because the operating license for a final waste disposal facility could be issued for determined duration, for 10 years at most, which – in case of meeting the operating conditions – can be extended for at least 10 years more on request.

The safety report can be partial or full scale. Full scale safety report should be prepared for the substantiation of the establishment, operating and closure license applications, also considering the results of the periodic safety reviews. Partial safety report should be prepared for the modification of certain – safety related – components of the facility, or for the modification of the licensed activity, for commencing the regulatory inspection and for finishing the active regulatory inspection.

Figure 1: **Safety cases within the life time of Bátaapáti repository [3]**



In the course of the geological research the best methods and technologies which are technically and economically attainable shall be employed and any data gathered or used in the research shall be stored in a uniform database. The parties conducting the research are entitled and obligated to use that part of the uniform database which is required for their work [4].

The elements of a typical structured procedure for the development and compilation of a safety case, at any particular stage of repository development, are the following [1]:

- The establishment of an assessment basis.
- The application of the assessment basis in a performance assessment.
- The evaluation of confidence in the safety indicated by the assessment.
- The compilation of the safety case.

The assessment basis is the combination of these three elements:

- The safety strategy.
- The system concept.
- The assessment capability.

The assessment capability represents the available resources, including the safety-relevant features, events and processes, assessments methods and models, site-characterisation data and other information (e.g. proper application of the methodology, models, databases and codes).

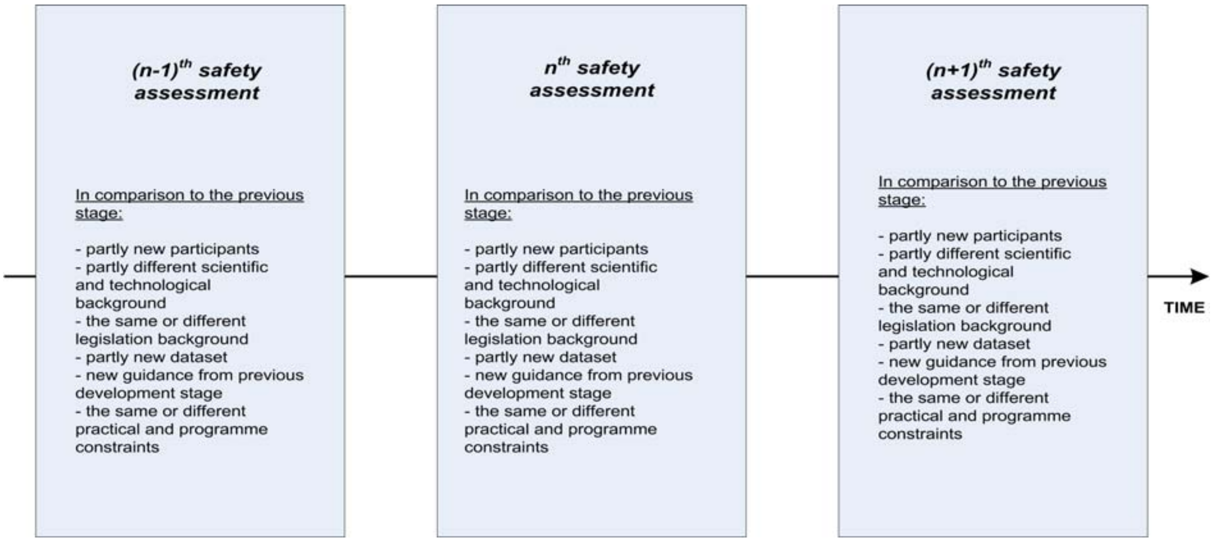
Every safety assessment development process begins with the re-evaluation and modification of the assessment basis. The re-evaluation and modification to the three elements of the assessment basis proceed concurrently because of the strong coupling between the elements – thus modification of the safety strategy or of the safety concept may guide to the significant changes in the assessment capability. But the assessment capability may need to be modified without the modification of the safety strategy and/or of the safety concept, too because of the change of decision-assessment environment. It means that the assessment capability depends on the decision and assessment environment, too.

Some more important elements of the decision & assessment environment are the following:

- People who participate in the decision-making and safety assessment processes.
- The scientific and technological background.
- Legislation background.
- Data (information) concerning the repository.
- Guidance from previous development stage.
- Practical and programme constraints.

Within each safety case development process different people participate, they have different scientific knowledge, different scientific/technological background (tools – e.g. codes, models, methodologies, instruments) and the legislation background of each safety development process may differ from one another.

Figure 2: **The change of decision-assessment environment**



The change of decision-assessment environment is rarely fast, thus it could be not obvious within a some-years long time frame, but it is evident within 40-50 years long perspective.

In the course of re-evaluation process of data concerning to the repository the guiding question which should be tried to answer: “is there enough information to make a defensible safety case?” [5] This re-evaluation process may conduct to data reduction as well as to data increase, too.

New information for the safety assessment may be produced mainly by the monitoring system after the preoperational phase of a disposal facility – theoretically. However, the amount of the data available for the safety assessment might be modified practically by the follows:

- Increasing of available data by monitoring system and the new or repeated measurements.
- Decreasing of available data by excluding data from the database of previous safety assessment stages.

The change of usable information for safety assessment is always induced by the change of decision and assessment environment.

In the course of re-evaluation data need to be excluded from the new safety assessments development process in general if the measuring circumstances (e.g. properties of the measuring instrument used to produce the data, including type, manufacturer, serial number, calibration data, measuring method, measuring range, measuring accuracy, dependence of accuracy on measuring range, when, why and by whom was the measurement made, whether any additive was used in the preparation phase of the measurement, if yes, how many additive was admixed to the medium need to be measured, etc.) are not verifiable. It means that data may be excluded from further works because of lacking some additional (auxiliary) information related to this data and the information that could make the repeating of previous data processing or the adoption of a new method possible.

The above mentioned issues concern not only the measured but the observed information, too. For example some results of geomorphological mapping are valuable for the experts only as long as these information might be verified in the outcrops. If these outcrops will disappear they will be destroyed and if the expert by whom these outcrops were written is not available any further, then the experts of the later generations will not be able to verify these information and will not be able to reprocess them from the same point of view as it was processed earlier. So the value of the information decreases in the expert's eye.

Most frequently the experts have difficulties with the verification of derived (calculated) data (e.g. mean as a result). If they are not able to determine which data were used for the calculation of the derived data (e.g. lack of traceability of the results, via chain of decisions and calculations, to their sources), they will not be able to use it in the new safety assessment process, and they will need to begin the data processing from the starting point again.

No law ordains that results originating only from the recent safety assessment process might be used to compile the new safety case, so the authorities might not refuse the license from the applicant referring to results originating from the previous safety assessment processes. But we are not allowed to exclude the possibility, that authorities will attempt to put pressure on the waste management company that the safety of repository ought not to be documented on the basis of several decade-old safety assessments results.

Environmentalist organisations and inhabitants from the neighbouring settlements may put similar pressure on the waste management company.

Similarly to the above mentioned, the experts are also averse to use several decade-old safety assessments results when compiling the new safety case so they might propose new investigational programmes. Their arguments will refer to meanwhile happened scientific and technological advance and they will keep quiet about that only trouble is they ought to use information issuing from the previous generations. This effort may be favoured by the legislation background, too (e.g. "In the course of the geological research the best methods and technologies which are technically and economically attainable shall be employed" [4]).

In my opinion it might be pointed out that data can easily lose their usefulness for the future generation if we do not retain, if we do not attach to them those information (metadata) which make their reproducibility, transparency and traceability possible. The lack of metadata can conduce to reduction of the assessment capability (by decreasing the amount of available data) and as a consequence some measurements, observation and calculation will be needed to repeat in one of the following safety assessment stages.

This lesson was learned by PURAM in the case of the last operation licensing process of Püspökszilágy surface repository: because of lacking some metadata related to the information used by the previous safety assessment stage (and also lacking a proper documentary method), practically almost all geoscientific investigation programme have been repeated.

The effect of the above mentioned occurrences may be reduced only if the following requirements in relation with the information used to the safety assessments will be fulfilled:

- Every measured, observed or derived information has to be built in the database together with their so called metadata describing the attributes of information (e.g. location- and time-co-ordinates, type of measuring tool, precision of measuring tool, name of the person who carried out the measurements, the applied measuring and/or data processing method, relations of data, and so on).
- Requirement of reproducibility, transparency and traceability need to be taken into consideration during the specification of metadata.

- These requirements need to be built in the data acquisition plan already in the very beginning of every project.
- The representative of waste management company has to participate always in the specification process of the metadata because the guarantee of the long-term considerations only thus are possible.
- Each information needs to be treated as individuality because it is not possible to find general rule for the specification of all metadata related to these information and needed by the future users.
- The relational database has to be preferred as a tool of information storage at present because a paper based information storage medium (e.g. reports) can appreciably reduce the chance of re-using of information (e.g. because there is difficult to gather the needed information from the reports or there is difficult to verify on the basis of reports whether all needed metadata have been recorded).
- Deletion of any information from the database has to be forbidden.
- The information (and/or metadata) upload into the database has to be a data-supplier's duty.

The Hungarian data-gathering practice does prove the above described requirements. To facilitate the specification of the needed metadata a recommendation is given by PURAM for the contracting parties. This recommendation imposes a minimal information content on the data-supplier that has to be guaranteed by the metadata. Some of these information are as follows:

- The place where data comes from (e.g. name(s), the type of location, co-ordinates, the type of co-ordinates and the accuracy of co-ordinates, etc.).
- The time/date of measurement, observation or derivation.
- Person, by whom the measurement (observation, derivation, data processing, verification, etc.) was carried out.
- The project, relating to the measurement, observation or derivation.
- Information about measuring tool (e.g. type, manufacturer, serial number, calibration information, accuracy, range, using method, measured parameter, measurement unit, etc.).
- The observation/derivation method (e.g. representation, related uncertainties, etc.).
- The raw data of derived information, related uncertainties.
- The related documentations in electronic form.
- The measured substance (e.g. water, rock, gas, plant, amount, volume, used chemical treatment, etc.).
- The geological formation and era when applicable.
- Links to the FEP when applicable.
- Etc.

In PURAM's practice we met the following difficulties and problems concerning the confidence till now:

- we have limited information about the practice of other countries in this scope so it is hard to say we are on the right track;
- users have various preconceptions, approaches about a database and feel;
- users have aversion from using and supporting a centralised database, everybody would like to use the own one;
- the complex structure of the database must be explored and accommodated by the users (without a good knowledge of database structure a researcher cannot use the database or able to form a good report request);
- users endeavour to make the identification of metadata formal, wherefore a strong supervision is needed from the waste management company;
- gathering and presenting all metadata for the research data originating from a previous safety assessment stage demands unusually many time and expenses, moreover there is impossible to determine in many cases;
- data quality assurance (to prevent filling up the database with useless or incorrect data.).

Proposal

It would be useful to compile a guide about this matter in the reflect of the international practice. Within the scope of this work it would be expedient to sum up the set of information used for the safety assessment, how these information should be arranged in groups, to determine metadata which are necessary for the long-term usefulness of these information in the safety assessment stages, and how the wrong determination of metadata can influence their reproducibility, transparency and traceability. PURAM is kind to participate in this kind of work because according to our current knowledge this may save us from needless expenses in the future.

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FOCUS ON ISOLATION AND CONFINEMENT RATHER THAN ON POTENTIAL HAZARD: AN APPROACH TO REGULATORY COMPLIANCE FOR THE POST-CLOSURE PHASE

B. Baltes, A. Becker, A. Kindt, K-J. Röhlig
Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH Köln, Germany

Abstract

In recent years, consequence calculations have been put in perspective in relation to other evidence used in a safety case. The authors believe that this development is a good and necessary one but that such calculations still have an important role to play in a safety case. There is, however, no single and recognised way about how to judge the regulatory compliance of calculation results in the regulatory process. The approach for the presently ongoing revision of Safety criteria in Germany is based on demonstration of the confinement of radionuclides. Calculated radionuclide concentrations in the accessible environment are seen as primary safety indicators which allow, together with other lines of evidence, to judge about the confinement capability of a repository system. Dose is not seen as indication of hazard to a hypothetical individual but as a means to judge about calculated concentrations.

Introduction: the revision of safety criteria in Germany

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

The Plan Approval Procedure (i.e. licensing procedure, "Planfeststellung") required by the Atomic Energy Act for federal installations for the safekeeping and final disposal of radioactive waste is in principle a one-step procedure which might last for the whole duration of a project. A stepwise approach is not explicitly implemented. The Plan Approval Procedure has a so-called "concentrating effect" for several fields of law. The safety criteria for the disposal of radioactive waste in a mine which were issued in 1983 [1] are tailored for a licensing situation at the end of a Plan Approval Procedure. Amongst the important cornerstones of the new waste management plan is a revision of these criteria. The revision is being carried out by GRS on behalf of the BMU. It has two objectives:

1. It is intended to update technical criteria according to the state of the art as described in the OECD/NEA's safety case documents as well as in the IAEA's Safety Requirements WS-R-4 "Geological Disposal of Radioactive Waste" [2]. This concerns especially the nature of the safety case as a collection of arguments for safety-comprising issues such as:
   - protection objectives;

– safety management;
– safety concept; and
– safety evaluation/assessment.

2. It is also desired to implement a stepwise approach where, at well-defined decision points, a safety case based on the knowledge achieved so far will be compiled, communicated to regulators and other stakeholders and be utilised to support decisions about how to proceed. Such a stepwise approach should form the basis for a process of constrained optimisation in accordance with ICRP and IAEA requirements. The implementation of such an approach in present legislation is, however, considered a challenge. Obviously, its implementation has implications for the formulation of the technical criteria mentioned above.

As requested in ICRP 81 [3], the optimisation process has to be carried out in an essentially qualitative way and is to be based on safety-relevant, technical, economic, planning, social and other objective functions. With regard to long-term safety, the radiation protection objective to limit the risk for an individual to sustain serious health damage due to exposure to radiation has to be met. In order to ensure this, optimisation is orientated on the objectives to achieve a high level of safety by means of a durable, complete and reliable confinement of the waste and of confidence into the evidence especially for post-closure safety. This paper focuses on the constraints for this optimisation process and on the ways assessment calculations should be utilised as one of multiple lines of evidence that the constraints are accounted for.

Constrained optimisation focusing on isolation and confinement

As explained above, the constraints focus on the safety functions of confinement and isolation. The constraints address four issues:

1. **Safety concept:** The complete and reliable isolation and confinement of the waste has to be ensured by a repository system where the main emphasis with regard to the assignment of safety functions is placed on the geologic barrier, i.e. on an isolating rock zone together with the geotechnical sealing components.
2. **Duration of confinement:** The confinement of the waste has to be ensured over at least 10^6 years.
3. **Completeness of confinement:** The confinement of the waste is considered to be complete if, for likely scenarios, only negligible amounts of contaminants will be released from the isolating rock zone, thus causing no adverse alterations of soil and water and no relevant risks for human beings and the environment.
4. **Reliability of confinement:** The likelihood of scenarios leading to higher releases than the ones mentioned above should be significantly smaller than 1.

Safety concept

The concept of an “isolation rock zone” stems from the work of the German Committee on a Site Selection Procedure for Repository Sites (AkEnd) [4]0. The isolating rock zone is defined as the “Part of the geological barrier which at normal development of the repository and together with the technical and geotechnical barriers has to ensure the confinement of the waste for the isolation period” [4].

The isolating has to be located in an area fulfilling the following criteria [4]:

- No large-area uplifts of more than 1 mm/a on average during the predictable period.
- No active fault zones in the repository area.

- No expected seismic activities exceeding Earthquake Zone 1.
- Neither quaternary nor expected future volcanism.

The AkEnd defined a number of minimum requirements for the isolating rock zone. These requirements concern:

- its field hydraulic conductivity;
- its thickness;
- its depth;
- its extension;
- the exclusion of risk from rock burst;
- the timeframe for which the minimum requirements on conductivity, thickness, and extension can be fulfilled.

In addition, a set of favourable properties of the isolating rock zone as well as of the overall geological setting were identified from which criteria and corresponding indicators were derived. Different weights were assigned to the criteria:

"**Weighting Group 1 – Quality of the isolation capacity and reliability of proof**

- no or slow transport with groundwater at repository level;
- favourable configuration of the rock formations, in particular of the host rock and the isolating rock zone;
- good spatial characterisability regarding the properties searched for;
- good predictability of the long-term stability of the favourable conditions.

Weighting Group 2 – Assurance of isolation capacity

- favourable rock-mechanic conditions;
- low tendency of the formation of water flow paths.

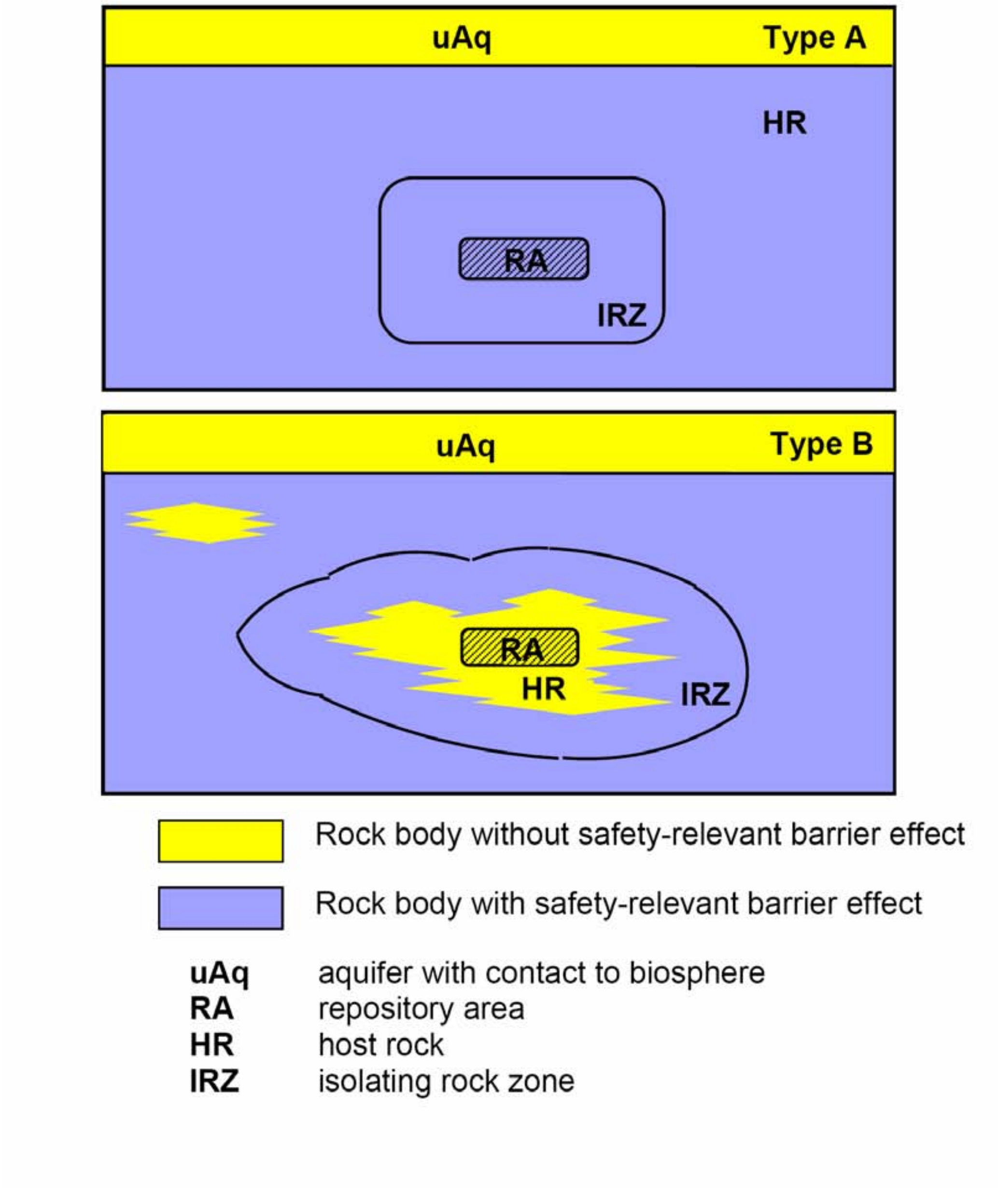
Weighting Group 3 – Other safety-relevant characteristics

- good gas compatibility;
- good temperature compatibility;
- high radionuclide retention capacity of the rocks;
- favourable hydrochemical conditions" [4].

The concept of the isolating rock zone should not be confused with the usual notion of the host rock for a repository. In contrast, a number of variants are conceivable for geological settings with an isolating rock zone (Figure 1).

The requirement of constructing a repository in an isolated rock zone implies a safety concept where the main safety functions should be provided by the geologic barrier together with the geotechnical sealing components.

Figure 1. **Main types of configurations between host rock and isolating rock zone Type A: Host rock is a safety-relevant part of the isolating rock zone Type B: Host rock is not a safety-relevant part of the isolating rock zone (from [4])**



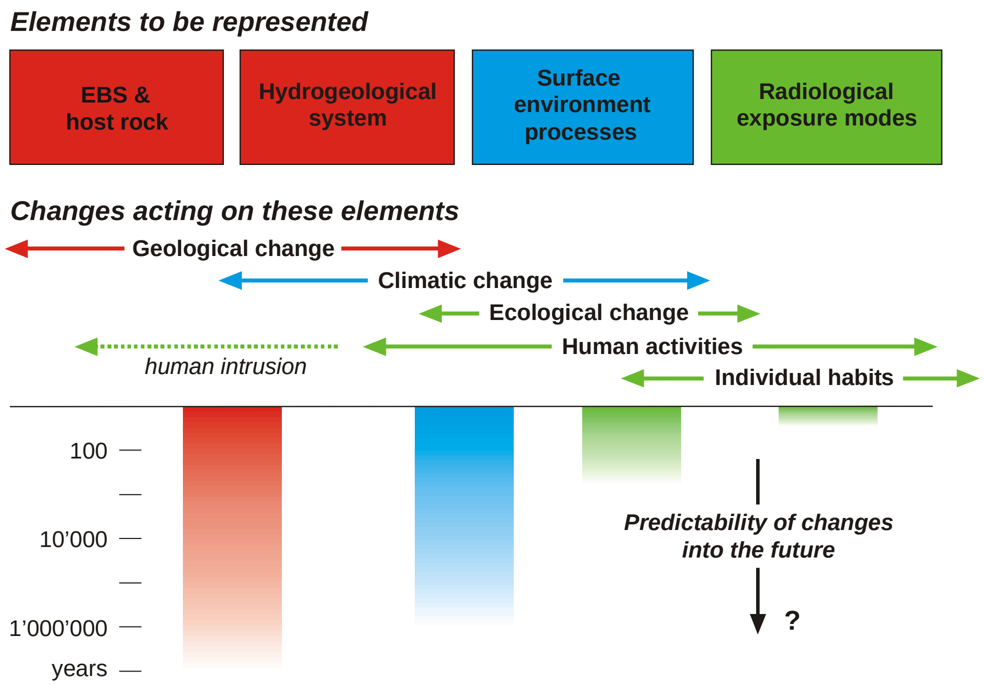
Duration of confinement

The AkEnd concluded that it is possible to identify sites in Germany the evolution of which can be predicted over at least 10^6 years and where an "isolating rock zone" as defined above can be found which will maintain its properties crucial for confinement (conductivity, thickness, extent, …) over this timeframe. Thus, this timeframe has been introduced as a constraint for the optimisation process and consequently also as an assessment timeframe. It is considered that optimisation becomes meaningless for times beyond the mentioned timeframe of predictability.

Completeness of confinement

Traditionally, the calculation of potential radionuclide release and migration and the evaluation of its radiological consequences in terms of calculated individual dose or risk were at the core of compliance assessments. In more recent years, considerable and sometimes controversial discussion took place about the value of such an approach, given the limited predictability of some of the system components to be modelled (Figure 2). The ICRP specified requirements about how to judge calculated doses or risks which are orientated on established concepts for radiological protection, but, at the same time, stated: "These estimates should not be regarded as measures of health detriment beyond times of around several hundreds of years into the future. In the case of these longer time periods, they represent indicators of the protection afforded by the disposal System." [3]

Figure 2. **Elements generally represented in safety assessment modelling, the changes that act on these elements and the impact on the predictability of the elements over time (from 0)**



In any case, the question remains whether accounting for the ICRP requirements indeed implies that the "ability of future generations to meet their needs and aspirations" is not compromised, as required in the Joint Convention [6]. Presently, it is also being discussed in the framework of the ICRP to which extent entities like calculated individual doses or risks, being orientated on the protection of human beings, are sufficient for judging the protection of the environment in general.

- Increasingly, assessment calculations are put in perspective in relation to other evidence used in a safety case. Accordingly, in the framework presented here such calculations are seen as one of multiple lines of evidence indicating that the constraints for post-closure safety are met. With regard to the completeness constraint it is considered that its assessment should as far as possible be based on indicators which can be calculated using modelling of components the evolution of which can be forecasted over the assessment timeframe (Table 1).

Table 1. **Indicators proposed for assessing post-closure safety (likely scenarios). Area shaded in dark grey indicates short time frame of predictability (compared to assessment time frame).**

Indicator	Location	Yardstick	Basis / motivation	Remarks/assumptions
Fraction of released amount of substance	Boundary of isolating rock zone	Percentage of amount of substance disposed of	Assessment of confinement	No direct relation to safety
Concentration of released U and Th	Boundary area of isolating rock zone	1 µg/l U, 0.1 µg/l Th	Alteration of natural concentrations	Restricted to Th and U (aggregation over isotopes, daughter products)
Contribution of released radionuclides to power density in groundwater	Boundary area of isolating rock zone	1 MeV per l porewater	Alteration of natural radioactivity	Aggregation over all nuclides, yardstick for impact on biota, but no dosimetry
Contribution to radiotoxicity flux in groundwater	Boundary of isolating rock zone	0.1 mSv/a	Alteration of natural radioactivity	Aggregation over all nuclides, "shortcut" of aquifer system, standardised "biosphere"
Nuclide concentrations in accessible groundwater	Near-surface aquifers	Specific per nuclide (^{238}U, ^{234}U, ^{226}Ra, ^{210}Pb, ^{235}U, ^{228}Th, ^{230}Th, ^{232}Th)	Alteration of natural concentrations	Restricted to natural radionuclides, no aggregation, present or plausible future hydrogeology, timeframe of predictability limited
Effective individual dose per year	Biosphere	0.1 mSv/a	Alteration of natural radioactivity	Aggregation over all nuclides, present or plausible future hydrogeology, standardised "biosphere", timeframe of predictability limited

Most of the suggested indicators (namely the fraction of released amount of substance, the concentration of released U and Th, contribution of released radionuclides to power density in groundwater, and the contribution to radiotoxicity flux in groundwater) are to be determined in the vicinity of the isolating rock zone. Other indicators (the nuclide concentrations in accessible groundwater and the effective individual dose per year) have a limited, confirmatory relevance. The yardsticks to be used are as far as possible orientated on conditions found in nature; radiological considerations should only be referred to for artificial radionuclides.

Reliability of confinement

Assessing the reliability of confinement is a challenge mainly for scenario development and assessment. For scenarios associated with potential releases exceeding the yardsticks mentioned above the likelihood of occurrence for these scenarios has to be assessed in a quantitative way and to be confirmed that this likelihood is significantly lower than 1. If such scenarios cannot be excluded from the assessment on regulatory grounds or because these scenarios are unlikely, their consequences should not exceed those from natural conditions, thus only causing a marginal additional risk for human beings and the environment.

Conclusions and future work

- In the proposed regulatory framework, assessment calculations serve as one of multiple lines of evidence substantiating that the optimisation respects the constraints concerning the safety function "confinement/isolation" which has to be ensured by the isolating rock zone.

- It can then be argued that, if the confinement of the waste is ensured, the protection objectives for humans and the environment are met. In turn, the confinement is ensured if the already existing system is perturbed as little as possible. This line of argument is being preferred in comparison to the utilisation of largely hypothetical biosphere models. The discussion about the possibility of "compromising the ability of future generations to meet their needs and aspirations" (Joint Convention, [6]) loses importance.

- The authors believe that the presented approach accounts for the often required, but less often implemented request to use safety and performance indicators additional to dose or risk. In fact, an increased role is assigned to these indicators which makes even the use of the terms "additional" or "secondary" questionable.

- As far as possible, indicators are relied upon which can be calculated based on modelling of components the evolution of which can be forecast over the assessment timeframe rather than on largely hypothetical biosphere considerations. For times beyond this timeframe optimisation becomes meaningless because prediction becomes impossible.

- As much as possible, the state of the undisturbed system serves as a yardstick to evaluate the indicators.

- This has, however, its limitations with regard to the assessment of potential releases of artificial radionuclides the assessment of which has to be based on radiological considerations based on standardised models.

This paper has to be seen as a snapshot of work which is still under progress. Amongst others, the following still open issues should be mentioned:

- Role of the indicators the predictability of which is limited in time but the use of which might be requested from the perspective of radiation protection (shaded in grey in 0) needs to be clarified.
- The indicators and corresponding yardsticks have been tested by means of scoping calculations for likely scenarios for repositories in salt domes and in indurated clay formations. Such testing is still needed for less likely scenarios and might result in changes concerning the chosen indicator set and / or the associated yardsticks.

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THE ENVIRONMENT AGENCY'S ASSESSMENT OF BNGSL'S 2002 POST-CLOSURE SAFETY CASE FOR THE LOW-LEVEL RADIOACTIVE WASTE REPOSITORY AT DRIGG

S.L. Duerden, A.J. Baker, D. Bennett, and I.J. Streatfield
Environment Agency of England & Wales

Introduction

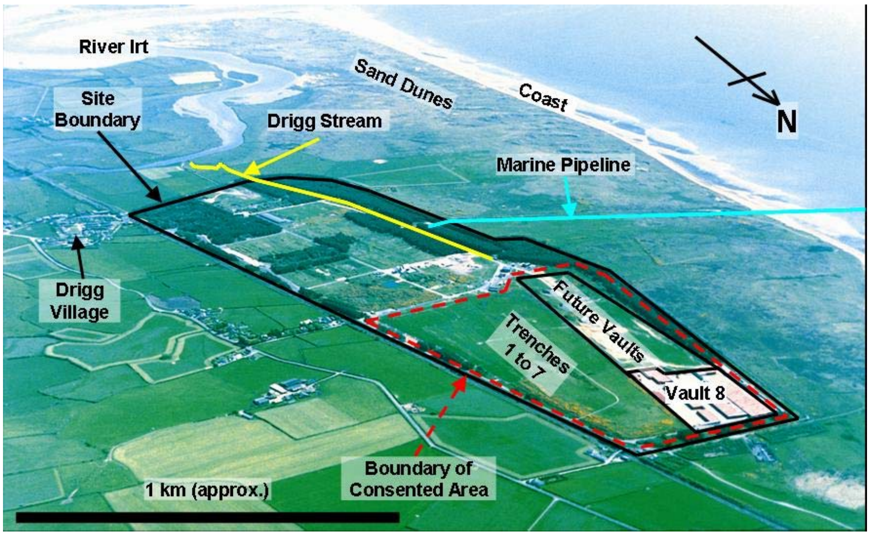
The Environment Agency regulates radioactive waste disposal in accordance with the Radioactive Substances Act 1993. British Nuclear Group Sellafield Ltd (BNGSL) is currently authorised to dispose of solid low-level radioactive waste (LLW) at a repository near the village of Drigg, in North-west England. We review authorisations for the disposal of radioactive waste periodically. We assessed BNGSL's 2002 Post-Closure Safety Case (PCSC) [1] for the LLW repository (LLWR) to inform the recent review of the LLWR Authorisation [2]. The paper presents an overview of our assessment of the 2002 PCSC for the LLWR, with particular emphasis on the process, rather than on specific review findings and recommendations (which can be seen at [3]). We have noted, in particular, some important lessons that are relevant to the consideration of any future applications for new near-surface or deep geological disposal facilities.

Background to the low-level waste repository (LLWR) at Drigg

An aerial view of the LLWR site is shown in Figure 1. The LLWR is the UK's only specialised disposal facility for low-level radioactive waste. The site was established in 1939 when an area of farmland was developed as a Royal Ordnance Factory. Radioactive waste disposal began in 1959. The wastes arise from all phases of the nuclear fuel cycle (e.g. uranium enrichment, fuel manufacture, power generation and reprocessing) as well as from industry, hospitals, universities, and defence facilities.

Between 1959 and 1995, approximately 800 000 m^3 of LLW was tipped into a series of seven shallow clay-lined trenches and covered with soil (Figure 1). Waste disposal to the trenches stopped in 1995 and the trenches were covered with a temporary cap that incorporates a waterproof membrane. Since 1988, waste has been compacted in steel drums. The drums are then placed in steel containers, similar to freight containers, and voids filled with a cementitious grout. The containers are then disposed of in a concrete vault (Vault 8 Figure 1). Waste items that are too big for the containers are grouted directly into the vault. The capacity of the existing vault is approximately 200 000 m^3 and it is nearly full. BNGSL plans to build additional vaults to accept a further 750 000 m^3 of conditioned waste. When waste emplacement at the LLWR is complete (which BNGSL suggest will be around 2050), and after a period of monitoring, a final cap will be constructed over the trenches and the vaults, and the site will be closed.

Figure 1. **Aerial view of the LLWR**



In November 2001, the UK Government and Devolved Administrations announced the formation of a new body having responsibility for the discharge of public sector civil nuclear liabilities, including those of BNGSL and of UKAEA. The body known as the Nuclear Decommissioning Authority (NDA) became operational on 1st April 2005 and took ownership of a number of sites, including the LLWR. The NDA will provide the driving force and incentives for systematically and progressively reducing the hazard posed by legacy facilities and wastes. It has a specific remit to develop an overall UK strategy for decommissioning and clean-up. BNGSL currently operates the LLWR under contract to the NDA.

Regulatory guidance

Principles and requirements

We regulate radioactive waste disposal in accordance with relevant laws, statutory guidance and Government policy. These considerations are described in detail in the Guidance on Requirements for Authorisation (GRA) [4] ,[1] and formed the basis of our review. We apply four general principles for the protection of the public when "reviewing the authorisations for future disposals to existing specialised disposal facilities":

Principle No. 1 – Independence of safety from controls

Following the disposal of radioactive waste, the closure of the disposal facility and the withdrawal of controls, the continued isolation of the waste from the accessible environment shall not depend on actions by future generations to maintain the integrity of the disposal system.

1. The GRA is currently under review. The three UK environment agencies will jointly be producing new guidance in two documents, one for deep geological disposal and one for near-surface disposal.

Principle No. 2 – Effects in the future

Radioactive wastes 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.

Principle No. 3 – Optimisation (as low as reasonably achievable)

The radiological detriment to members of the public that may result from the disposal of radioactive waste shall be as low as reasonably achievable, economic and social factors being taken into account.

Principle No. 4 – Radiological protection standards

The assessed radiological impact of the disposal facility before withdrawal of control over the facility shall be consistent with the source-related and site-related dose constraints and, after withdrawal of control, with the risk target.

The GRA describes a number of more specific requirements underpinning these principles. We take account of these requirements when reviewing authorisations for existing facilities, but we do "...not *seek to apply the more specific requirements of the GRA to historical disposals at existing facilities where the standards adopted at the time of disposal were significantly different*" (GRA paragraph 1.14).

The GRA Principles and Requirements provided an invaluable framework for our review of the PCSC. However, we needed to consider and discuss with BNGSL how we would apply the detailed requirements to an existing facility with past (or "historic") disposals that were authorised in accordance with previous regulatory standards. This is an important point to consider when regulating a new facility since regulatory decisions will be made over a long period, during which regulatory standards may change and, at the time of closure, all disposals to the facility would be "historic". The GRA currently states that *"disposals will not be regarded as complete until all the requirements of the safety case have been met, including sealing and closure of the facility"*. We will consider the adequacy of this statement when we review the GRA.

Early dialogue with the operator/developer

We engaged in regulatory dialogue with BNGSL prior to and during the development of the 2002 PCSC (late-1996 to September 2002) [e.g. 5,6,7]. Much of our work was based on information classified by BNGSL as "*interim work in progress*". This information was not accessible to the public, although it has since been made more widely available by BNGSL. In general, such confidentiality requirements are related to commercial or security considerations. We would strongly urge an operator/developer to limit as far as possible the quantity of information that is not publicly available. A balance may need to be made between providing enough information to promote public acceptance while at the same time not releasing sensitive information.

Interim information from BNGSL tended to focus on specific scientific and technical issues. It did not provide an overall understanding of the safety issues, nor a rounded safety case. In response, we provided detailed and site-specific guidance on our expectations of a PCSC to meet the requirements of the GRA, in the form of Technical Review Criteria (TRC) [5,6,7]. As a consequence, we generated a large number of regulatory issues that we were unable to prioritise in terms of the potential impact on long-term safety. At this stage, it was important not to unduly constrain BNGSL's options during the development of the safety case. We also had to avoid compromising future regulatory decisions.

Despite these problems, we consider that this early dialogue and the use of technical review criteria led BNGSL to undertake a more appropriate range of studies, to increase its understanding and to provide an improved basis for the PCSC. It also enabled us to focus regulatory review procedures, establish an appropriate review team, and develop and test assessment tools. However, these benefits were realised at considerable cost, albeit spread over a number of years. On balance, we consider that it is essential to have early dialogue with the operator/developer. This should ensure a common understanding of requirements, pertinent safety issues, and the approach to the development of a safety case.

Future developments

The GRA was written with a new, deep geological disposal facility in mind. Despite this we found that it provided a valuable framework for assessing an existing shallow/near-surface disposal facility. We consider there is a good case for producing separate guidance for deep geological disposal and for shallow disposal, as there are significant design and technical differences between these sorts of disposal facility. We are currently progressing this. Our assessment of the PCSC focused on the future radiological impacts of the LLWR. Non-radiological environmental impacts are not covered specifically in the GRA, but were considered as part of the Authorisation Review. Our review was also undertaken on the premise that the standard of environmental control needed to protect man will ensure that there is no undue impact on other species. Our review of the GRA will consider what further guidance related to non-radiological impacts and effects on non-human species might be appropriate.

Review and re-assessment

We will continue to regulate disposals to the LLWR throughout its operation and closure, until control is withdrawn. We expect the safety case for the LLWR to be revised throughout this period. In particular, we would expect any proposed developments to be considered, taking account of their potential impact on post-closure safety. In this respect, it is worth highlighting the need to establish a clear understanding of the links between decisions made during the development and operational stages of a disposal facility and post-closure safety.

A safety case should be reviewed and revised on a regular basis. This has significant resource implications for both the operator/developer and regulators, not least in the need to maintain essential expertise and experience.

Clearly, there is a need to establish a means for taking account of future developments within the safety case (such as, new or imminent statutory requirements, developments in science and engineering; or in safety assessment; waste treatment; and packaging).

There is a need for ongoing dialogue between the operator/developer and the regulator, in order to prioritise issues.

Design, construction and submission of a safety case

The overall document structure for the PCSC was as follows:

Level I: Overview Report [1].
Level II: 15 “Supporting Reports” [8].
Level III: over 200 “Underpinning References”.
Level IV: many hundred Project Information Documents.

We expected the safety case to be made and supported at the top two levels (LI and LII). Therefore our intent was to focus the review of the PCSC principally on the Overview Report and the Supporting Reports, and to consider review of relevant Underpinning References only where more detail was required on any key safety issue [9]. However, in practice, we needed to review the more detailed information contained in the Underpinning References, in order to assess adequately the validity of the assumptions and data comprising the safety case. In a few cases, this also required examination of Project Information Documents. BNGSL provided the PCSC at the end of September 2002, and underpinning references and other project information documents cited in the 2002 PCSC during late-2002 and the first half of 2003. During 2003 and the first half of 2004, BNFL supplied additional information in order to support the 2002 safety cases. Such piecemeal receipt of key information seriously impeded our review. For any future assessment we would aim to agree, at the outset, the nature of the information to be provided by the operator/developer in order to plan the scope of our work, and to ensure that all necessary information is available.

Understanding and managing the construction of a safety case is a key task for the operator. There can be a tendency to direct effort disproportionately towards managing the individual scientific and technical pieces of work (e.g. developing modelling tools) and then trying to fit these various and disparate components together, rather than understanding and putting the key arguments and evidence together to make a safety case. For this reason we would recommend an initial top-down approach to building a safety case rather than bottom-up (i.e. noting the safety objectives, defining the major components of the case, and with subsequent consideration of detail where this is required).

Current UK Government guidance [10] for environmental risk assessment and management describes an iterative approach to identify, screen, prioritise and quantify risks. The level of effort put into assessing a risk should be proportionate to its priority (in relation to other risks) and complexity (in relation to understanding the likely impacts).

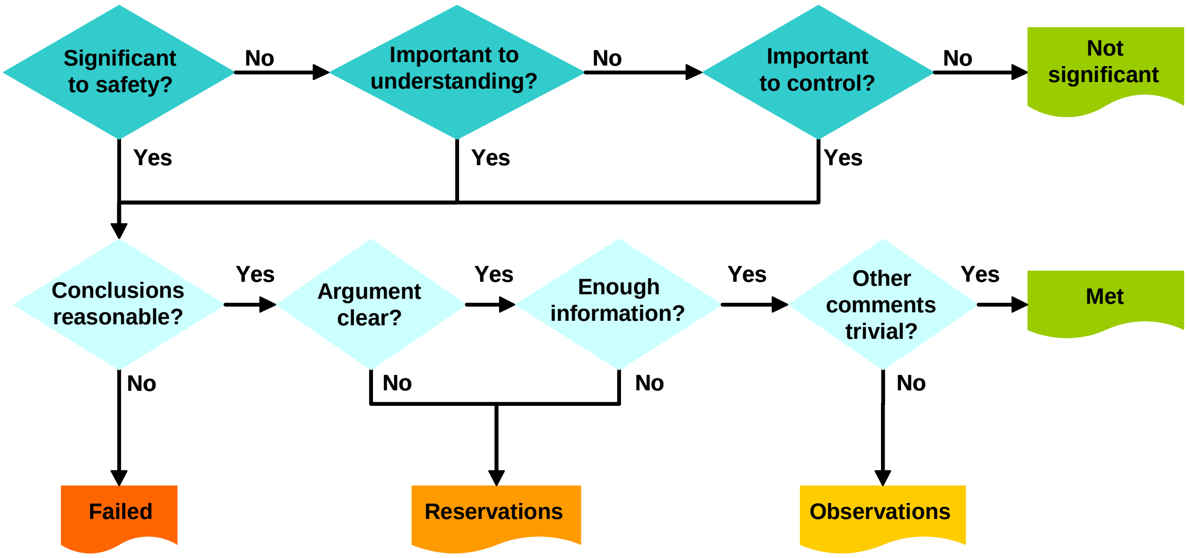
The regulatory review process

Focusing on issues of significance (How good is good enough?)

Our assessments of the PCSC were criterion (or issue) driven; discrete reviews of individual documents were not undertaken. The review team members assessed the PCSC against specific review criteria derived from the GRA. We conducted our assessments in accordance with a procedure for determining the significance of issues (Figure 2), in order to focus effort on issues of significance to repository performance and risk management. Where a reviewer considered that BNGSL's conclusions were not reasonable, that the line of argument was not clear, that sufficient information was not available, and/or that remaining review comments were not trivial, one of the following actions was recommended:

- Request further information from BNGSL to support its position.
- Suggest more detailed assessment of the issue by the Environment Agency.
- Suggest further work by BNGSL.

Figure 2. **Assessing the significance of issues during the safety case reviews**



There can be a tendency for an operator to present the full package of safety-related information without any attempt to pull together the pertinent information to make and support a safety case. This should not be left for the regulator to do. It is the regulator's primary task to identify any significant gaps or weaknesses in the safety case. It is difficult for the regulator to do this effectively if the safety case has not been assembled and documented.

It is of central importance for operators and regulators to understand the important aspects of a safety case, including the aspects of design and operation that really matter. For this reason, a safety case must be presented as clearly and simply as possible. The regulators task is to ensure that safety arguments are fully justified and supported. In this respect, it would be helpful for the safety case document to be as specific as possible in references to information elsewhere.

We provided general guidance on how we would use quantitative risk estimates to guide our assessment of the PCSC and what we would expect from an operator/developer in order to demonstrate compliance with a risk target [11]. It is difficult to provide more direct guidance without reference to a specific safety case. In very general terms, the level of detail required from the operator and the level of regulatory scrutiny should be proportionate to the risk and complexity [10], and to the underlying uncertainties. Regulatory judgements will be made on the robustness of the safety case arguments, including amongst other things:

- the confidence that can be placed on quantitative estimates of risk;
- the identification of a wide range of uncertainties and consideration of their effects on overall system performance. Effort should be focused on constraining and, where appropriate, reducing uncertainties that are significant with respect to safety;
- the clarity and transparency of, and justifications for, assumptions made in the safety case, and the consideration of alternative assumptions.

Dialogue during the review process

It is necessary to define the nature and extent of dialogue between the regulator (and any supporting external experts) and the operator during the review of a safety case. When we reviewed the PCSC we did not, for example, allow reviewers to clarify their understanding through dialogue with BNGSL because we considered that a safety case should stand on its own and be understandable without recourse to dialogue with the operator.

It is also necessary to establish the nature and extent of dialogue with statutory consultees, public bodies and government and the wider community such as Non-Government Organisations (NGOs), local authorities, and the general public. For our review of the PCSC we chose not to communicate proactively with other parties, although we did make the process as transparent as possible by putting key documents on the Public Registers and raising awareness that we were undertaking the review.

Building confidence in the quality of regulatory judgements

Our assessment of the PCSC was conducted in accordance with a clearly defined plan and guidance for reviewers [9,12]. It involved a team of experts comprising Environment Agency staff and consultants from industry and academia. Thorough review of a PCSC requires detailed understanding of a number of scientific and technical disciplines. A wide range of expertise is required, ideally with an understanding of the regulatory and radioactive waste perspectives. This of course leads to problems in establishing and maintaining commitment to a process that can span a long period of time.

Where expertise is not available within a regulator's organisation, external individuals or organisations might be commissioned to contribute to the review. It is important that such individuals or organisations are sufficiently independent. We established a general conflict of interest criterion at the beginning of the process.

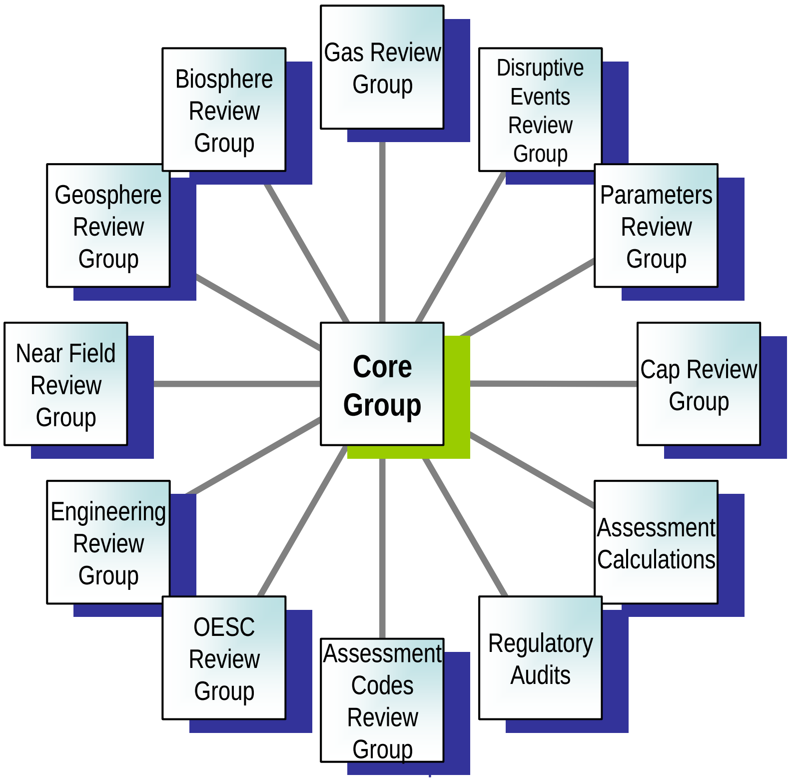
"The organisation or contracted expert should not have undertaken, or will not undertake, any relevant work on behalf of the operator which would compromise the regulator's independence".

Finding truly independent individuals or organisations is becoming increasingly difficult. This issue will need to be considered carefully in assessing any future application for disposal and will require decisions on the suitability of key individuals to be made on a case-by-case basis. It would also be worth remembering the extensive expertise available within other national regulatory organisations.

The regulator needs to consider how it will take ownership of the views of external experts, particularly when those views might be used as a basis for regulatory decisions. Our assessments of the PCSC were conducted by groups of reviewers drawn from the Review Team (Figure 3). Importantly, each Review Group included at least one Environment Agency staff member to ensure understanding and ownership of the review outputs and to focus on regulatory issues.

The work of the Review Team was overseen by a Core Group. The purpose of the Core Group was to identify and assess key issues and those that cut across scientific disciplines, prioritise areas identified for further assessment work, and make recommendations concerning the future of the LLWR and any necessary authorisation conditions. Inclusion of the Environment Agency's Site Regulator for the LLWR was essential in this respect.

Figure 3. **Review groups and supporting studies conducted by the Review Team**



Recording review comments and evolving positions

The review process may take place over many years, during which there may be significant organisational change. It is important to retain a memory of the basis of regulatory decisions taken throughout the review process. It is, therefore, essential for regulators and operators to establish and implement a well thought out and documented process for managing information over many years. We developed an issue assessment process and associated issues database [13,9]. Each issue is documented on an Issue Assessment Form (IAF), which provides specific comments on the PCSC on the basis of stated regulatory principles and requirements. The Issues Database records all dialogue relating to the developing safety case (1996 to 2002) and to the 2002 PCSC. It enables rapid retrieval of key information.

Use of review outputs

At an early stage in the review process we identified a need to have a clear idea of how outputs from our assessment of the PCSC would feed into the regulatory decision-making process [11,14]. Not all comments, and/or recommendations had the same degree of significance. Some required urgent resolution and fed directly into authorisation requirements. Some may be subject to ongoing dialogue. Some comments from our assessment of the 2002 PCSC may be superseded as a result of future developments. Some recommendations may require actions by ourselves and other parties.

Regulators need to consider how to communicate review findings to a wide range of audiences. Maintaining the detail while at the same time ensuring the clarity of key messages may mean there is a need for different documents aimed at different audiences. However, in this approach, there is the danger of being considered to be manipulating or withholding information. Regulators should ensure that outputs adequately and accurately reflect the regulatory position and should limit the potential for alternative interpretations. This requires careful scrutiny of final outputs, and the resources required for this task (which can be considerable) should not be underestimated. We found it helpful to arrange for scrutiny of our outputs by a member of staff who was not involved in the overall process.

Closing comments

BNGSL has invested a huge amount of effort on the tools and work packages that underpin the PCSC. Similarly, the Environment Agency has spent considerable resources in assessing the PCSC and the associated development programme. The PCSC for the LLWR is a valuable piece of work. In reviewing the PCSC, the Environment Agency has not only been able to evaluate the safety arguments for a specific facility, but has also gained valuable experience in the approach to the review of such a post-closure safety case. A key task now is to ensure the lessons learned are applied when we review any future application or authorisation for the disposal of radioactive waste.

For the LLWR, we are requiring a range of improvements to the safety case. We intend that these improvements will contribute towards the production of a final safety case that will provide a satisfactory basis for decision making.

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Quality Assurance.
Engineering Design.
Engineering Performance Assessment.
Site Development Plan.
Inventory of Past and Potential Future Disposals.
Geological Interpretation.
Hydrogeological Interpretation.
Far-Field Geochemical Interpretation.
Near-Field Biogeochemistry.
Software Tools and Codes.
PCRSA Approach.
PCRSA Scenarios and Calculation Cases.
PCRSA Process System Analysis.
PCRSA Results.
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PERSPECTIVES ON DEVELOPING INDEPENDENT PERFORMANCE ASSESSMENT CAPABILITY TO SUPPORT REGULATORY REVIEWS OF THE SAFETY CASE

J.R. Winterle
Center for Nuclear Waste Regulatory Analyses; U.S.

A.C. Campbell
U.S. Nuclear Regulatory Commission

Abstract

The U.S. Nuclear Regulatory Commission (NRC) has the responsibility under U.S. statutes and regulations to conduct a review of the U.S. Department of Energy (DOE) license application for a potential high-level nuclear waste repository at Yucca Mountain, Nevada, USA. A key component of the DOE license application will be a total system performance assessment to demonstrate compliance with regulatory requirements for postclosure repository performance. The NRC staff, with assistance from the Center for Nuclear Waste Regulatory Analyses (CNWRA), has developed an independent performance assessment review capability that includes applying available risk insights to be able to identify those aspects of the DOE post-closure safety case that are most important to demonstrating regulatory compliance. To advance the development of risk insights, the NRC and CNWRA staffs have developed an independent total system performance assessment model of the potential Yucca Mountain repository. This paper provides an overview of perspectives gained from the development and iterative improvement of independent total system performance assessment capabilities and how those capabilities will aid in the regulatory review of the DOE performance assessment.

Introduction

The United States is considering the Yucca Mountain site in southern Nevada as the location for a geologic repository for disposal of high-level radioactive wastes. The DOE is charged with the tasks of site characterisation, repository design, and development of a license application to demonstrate that the potential repository will meet regulatory requirements specified in Part 63 of Title 10 of the Code of Federal Regulations (10 CFR Part 63). DOE is presently preparing its license application for a geologic repository at the Yucca Mountain site. The U.S. regulations require that reasonable expectation of safety be demonstrated in the license application for post-closure performance. Moreover, the U.S. approach envisions a safety case will be documented in the license application. Thus, in this paper, the use of the term safety case should be considered synonymous with the license application. The NRC will review a DOE license application to ensure the regulatory requirements of 10 CFR Part 63 are met. The DOE has stated that, in June 2008, it will submit a license application for NRC review. Under the Nuclear Waste Policy Act of 1982 as amended, the NRC has three years to issue a final decision approving or disapproving the issuance of a construction authorisation. That decision will not be made until the NRC completes its review process, including an acceptance review, a safety evaluation, and adjudicatory hearings. During the time period prior to receiving the license application (the "prelicensing period"), the NRC staff is reviewing the DOE site characterisation activities and investigations to help identify potential licensing issues and prepare the NRC to conduct its review.

A key part of the DOE safety case (i.e. DOE license application) will be a total system performance assessment (TSPA) to demonstrate compliance with regulatory post-closure performance objectives. Because the TSPA is important to the DOE license application, NRC prelicensing activities have placed high priority on developing an understanding of how site characterisation and design information are incorporated into the DOE TSPA model. To that end, the NRC and its technical support organisation, the Center for Nuclear Waste Regulatory Analyses (CNWRA), have developed an independent performance assessment review capability that includes application of risk insights to help identify those aspects of the DOE safety case that are important to demonstrating regulatory compliance.

To advance the development of risk insights, the NRC and CNWRA staffs have developed an independent total system performance assessment (TPA) model of the potential Yucca Mountain repository. This independently-developed model provides NRC with a tool for understanding the relative importance of various natural and engineered barrier system and subsystem components without having to wait for, and rely solely on, DOE models and data to develop this understanding. While there are clear benefits to having this independent performance assessment capability, there are also costs in terms of resources and staff time. This paper provides an overview of perspectives gained from the development of independent total system performance assessment capabilities and how those capabilities will aid in the regulatory review of the DOE performance assessment. It is hoped that these perspectives benefit regulators of other national or international high-level waste disposal programmes who may be considering developing their own independent performance assessment capabilities.

Purpose of performance assessment

To begin, it is important to understand that license applicants and regulators have decidedly different motivations for developing their TSPA models. In the U.S. programme, the DOE, as a potential license applicant, is required by regulation (10 CFR 63.113) to conduct a performance assessment to quantitatively estimate radiological exposures to the reasonably maximally exposed individual at any time during the postclosure compliance period. The performance assessment also is appropriate for demonstrating compliance with performance objectives for groundwater protection and human intrusion and for providing the required demonstration of multiple barriers. Specific requirements (10 CFR 63.114) are that the applicant's performance assessment must (i) include data related to the natural and engineered barrier systems used to define parameters and conceptual models; (ii) account for uncertainty and variability in parameter values; (iii) evaluate the effects of the alternative conceptual models that are consistent with available data and scientific understanding; and (iv) provide the technical basis for models used in the performance assessment. Thus, the purpose of the applicant's performance assessment is to meet regulatory requirements to provide a defensible demonstration of compliance with performance objectives.

Conversely, there is no requirement for the NRC, as regulator, to develop a performance assessment. It is the applicant's performance assessment that will form the basis for determining whether compliance with performance objectives has been demonstrated. An important purpose of independent performance assessment for the regulatory agency, however, is to develop risk insights to improve understanding of which processes, parameters, and alternative conceptual models are most important to repository performance. In addition, NRC and CNWRA staffs have gained important insights from the process of developing a complex performance assessment, such as the complex manner in which interrelations between various parameters and model uncertainties can affect estimates of potential repository performance.

A performance assessment developed by a regulator can be used to develop a better understanding of performance assessment issues through examination of alternative models and data sets. For example, a regulator may question whether certain conceptual models, inputs, or assumptions that may be included in a potential applicant's performance assessment model are appropriate given available information. In this case, the regulator could notify the potential applicant that additional technical justification or an alternative analysis may be warranted, and the potential applicant may agree to provide such additional information. If the regulator has independent performance assessment capability, the regulator can use its own performance assessment model to analyse alternative approaches or bounding parameter values to assess the degree to which an alternative approach might affect performance. If the regulator's performance assessment analyses suggest that a particular technical issue is unlikely to affect potential repository performance, it may not be necessary for the potential applicant to submit a detailed justification on a technical issue that is of little overall importance. Conversely, if analysis indicates a technical issue may be highly significant to performance, the regulator will have a stronger basis for requesting that the potential applicant provide additional technical justification or adopt a revised approach.

Also to be considered are the complexity and level of uncertainty in site characteristics and design parameters. If, during the pre-application phase, a regulator has enough information to conclude that a particular site is sufficiently characterised and understood and that the design is sufficiently robust, there might be little need to develop independent modeling capability. A regulator might also choose to forego independent model development if sufficient information is provided by a potential applicant, and the regulator considers that the site conceptual model, modelling approach, and treatment of uncertainty are sufficient to enable the regulator to perform a review. A regulator would likely not gain additional risk insights by developing an independent model based on the same conceptual approach and treatment of uncertainty as contained in a potential applicant's model.

Considerations during development of independent performance assessment capability

Level of detail

For regulators who decide to proceed with development of an independent performance assessment model, budgetary and staff constraints will often necessitate development of models that are relatively simplified compared to a model developed by an applicant to support a safety case. This is not necessarily a problem or drawback. Since the regulator's goal is to develop a model that can provide risk insights with a total system context, a simplified model may be better suited to this purpose and permit more rapid adaptation for exploring alternatives and technical uncertainties. For example, the TPA code developed by NRC and CNWRA [1] uses a series of one-dimensional stream tubes to model radionuclide transport in the geosphere, whereas the most recent DOE model uses a fully three-dimensional particle-tracking approach to model radionuclide transport [2]. While the more complex DOE model incorporates a more detailed representation of the site geologic characteristics, the geologic representation is based on deterministic interpretations that can be difficult to update or modify to evaluate effects of geologic uncertainties on flow and transport. Conversely, the simplified one-dimensional approach used in the TPA model can easily be used to evaluate such uncertainties simply by varying or stochastically sampling input parameters related to transport distances and transport properties of geologic layers.

A regulator may also choose to include features, events, or processes in its independent model that may not be included in the model that the applicant is developing to support its safety case. For the regulator, evaluation of processes that may not be considered by the applicant in developing its safety case provides a useful method for evaluating the relative importance of alternative conceptualisations. It is important to note that the inclusion of alternatives in the regulator's model does not necessarily mean the regulator advocates or will require the same alternative conceptualisations to be part of the applicant's model. The insights gained from the alternative models, however, can provide the regulator with a basis for focusing its reviews on the justification provided by the applicant for excluding certain features, events, or processes.

Use of conservatism in the treatment of uncertainty

Abstracting complex processes into a performance assessment framework generally necessitates the use of simplifying assumptions and approximations that may not fully represent all aspects of the complex nature of a geologic repository system. Model developers, however, cannot ignore the need for conceptual models, model abstractions, and input parameters to represent essential features and key processes and to be technically defensible. One means of representing simplified model abstractions and approximations is to demonstrate that chosen approaches are conservative – that is, they bound performance relative to what more realistic and detailed physical representations would yield. While the use of conservatism in performance assessments is a generally accepted practice, there are several factors that need to be considered in applying this approach.

First, it is not always clear when a particular assumption or parameter estimate is a-priori conservative. Selection of a conservative parameter value or conceptual model based on process model assumptions may not necessarily lead to a conservative result for overall system performance because of non-linear interactions in the system model. In some cases, a more realistic parameter set or model may be necessary to capture the complexity of the system to the extent permitted by the state of the art in modeling, available data, and relevant uncertainties; or, it may be necessary to evaluate more than one simplified model abstraction (e.g. evaluating both high and low bounding estimates for the parameter) to ensure the effects on overall system performance estimates are sufficiently understood. Thus, regulators should attempt to evaluate system-level effects in adopting simplified assumptions that are thought to be bounding or conservative.

A second consideration is that the use of excessive conservatism may bias results in a manner that obscures the overall risk significance of other processes or parameters that are treated more realistically. This can make it difficult to evaluate uncertainties and sensitivities of system performance to specific parameters or models. For example, Mohanty and Nes [3] investigated the effects of using broad ranges of input parameter distributions on the identification and ranking of influential parameters to illustrate how rankings may change depending on the spread of input distributions. By evaluating parameter values that were substantially higher or lower than an assumed "realistic" value, they showed that, over the uncertainty range of a parameter, the model output sensitivity to the parameter could vary from being insensitive to highly sensitive, depending on the "level of conservatism" assumed. During iterative performance assessment development, it is important that the relative importance of modeled processes is properly understood so that future model iterations can be focused on the processes most important to the demonstration of safety. Because simplifying assumptions in complex safety analysis models is typically unavoidable, analysts should evaluate the effects of the assumptions or use alternative parameter ranking methods, such as component sensitivity analysis [4], to ensure that influential parameters are correctly identified.

A third factor to consider before adopting a conservative approach is the potential compounding effects on performance estimates when multiple conservative approaches are adopted. A truly conservative model abstraction or input value will produce system-level performance estimates demonstrably less optimistic than would be obtained from more realistic approaches. Compounding conservatisms, therefore, make performance estimates incrementally more pessimistic and may lead to conclusions about overall system performance that are inconsistent with real physical processes. Regulators should carefully consider the combined effects of conservative assumptions and strive as much as possible to improve realism in treating of uncertain processes and parameter estimates. In some cases, more realistic models may be needed to avoid overly conservative simplifications; in other cases, additional information may be warranted to obtain more defensible parameter estimates instead of using conservative bounding values or wide uncertainty distributions.

Role of independent performance assessment capability in developing risk insights

Application of risk insights to regulatory reviews

A regulator's risk insights are drawn from the process model analyses, information, and performance assessment analyses developed by a potential license applicant, and are enhanced by independent technical analyses and through the development and exercise of its own independent performance assessment tools. The risk insights provide a basis for focusing regulatory reviews on those aspects of a potential safety case that are most significant to waste isolation. A recent example of the NRC staff's understanding of risk information for the potential Yucca Mountain repository system is described in the Risk Insights Baseline Report [5]. This report groups risk insights for various features and processes into three categories of relative significance (high, medium, and low), based on contribution to, or effect on, the waste isolation capabilities of the repository system.

In addition to these risk insights, NRC also will consider the important barriers identified by DOE in their license application. NRC will give a high level of review focus to those aspects of the repository system identified by DOE in its license application as barriers important to waste isolation and also identified by the NRC as having a high significance to waste isolation. A high level of focus would also be given to those aspects of the repository system where DOE has identified those materials, structures, or features of the repository system as barriers important to waste isolation, but where NRC has identified that the comparable aspects of the repository system may have low significance to waste isolation. The NRC staff review would apply a moderate focus to areas where it has identified barriers as having high significance to waste isolation, but where the DOE has not identified the comparable materials, structures, or features as barriers important to waste isolation.

An example of this decision construct is described in Table 1. NRC uses a risk-informed approach to focus review efforts on the multiple barriers that may be put forward by DOE in its licensing case [6]. A more thorough discussion of how risk insights can be applied to a licensing review is provided by Leslie [7].

Table 1. **Risk-Informed Logic for Determining the NRC Staff's Focus in its Review [7]**

NRC Relative Significance	Identified by the DOE as Barrier	Level of Focus in NRC Review
High	Yes	High
High	No	Moderate
Low	Yes	High
Low	No	Low

Risk insights gained from independent performance assessment analyses

The Risk Insights Baseline Report [5] provides numerous examples of how independent performance assessment analyses can help to develop a better understanding of barrier capabilities. Below we focus on a single example of an evaluation of barrier capability that illustrates how risk insights can be developed and used, not only to focus the regulatory review process on risk-significant uncertainties, but also to guide the regulator in refining model abstractions in subsequent iterations of an independent performance assessment process. The example relates to the persistence of a passive oxide film on the surface of the waste package; this potential barrier was identified in the Risk Insights Baseline Report [5] as having high significance to waste isolation. The presence of a passive oxide film on the surface of the waste package outer barrier (consisting of Alloy 22) is anticipated to result in very low general corrosion rates of the waste package. Under environmental conditions where a stable oxide film is maintained, corrosion is uniform and occurs at a slow rate. The risk ranking for this process was based in part on an analysis using the independent total system performance assessment code TPA Version 4.1j [1]. This performance assessment analysis [5] compared the calculated expected dose for two cases: (i) a base case, assuming passive conditions persist for all waste packages and (ii) an alternative case, assuming passive conditions are not maintained for 25 per cent of the waste packages. As a result of the low corrosion rates under passive conditions for the base case, the predicted waste package failures from corrosion occurred at times ranging from 37 000 years to 403 000 years; the expected dose within 10 000 years was driven only by releases from assumed initial defects. For the alternative case, the absence of passivity was assumed to result in uniform corrosion rates that resulted in waste package failure times ranging from 400 to 4 000 years; the calculated expected dose was approximately 100 times greater within 10 000 years. A similar performance assessment analysis presented in the Risk Insights Baseline Report [5] concluded that the mode of corrosion failure (e.g. localised corrosion, uniform corrosion, or stress corrosion cracking) also can affect expected dose estimates because the failure mode affects the size of the opening that may allow water to enter a waste package. This analysis was done by limiting the amount of water assumed to enter waste packages exhibiting localised corrosion and not applying any limitation on waste packages affected by general corrosion. Based on this analysis, the corrosion failure mode was judged to be of medium significance to waste isolation. Given the high potential significance of passive oxide film persistence on the waste package surface and using the example of the decision logic in Table 1, NRC would assign a high or moderate level of focus on this issue during a licensing review, depending on whether DOE will take credit for this process as a waste isolation barrier component.

NRC and CNWRA also apply such risk insights for helping to determine when additional independent analyses may be conducted during the prelicensing period to improve understanding of the related uncertainties. For example, CNWRA conducted independent laboratory experiments [8] on Alloy 22, the proposed waste package outer barrier material, which confirms the assumption of slow corrosion rates so long as the passive film persists. These experiments also showed, however, that localised corrosion of Alloy 22 could occur in environmental conditions characterised by oxidising or acidic, concentrated chloride containing solutions with low concentrations of inhibiting oxyanions. The results of these independent analyses then formed the basis for a revised performance assessment abstraction that is capable of explicitly evaluating uncertainties in the near-field environmental and chemical conditions that can lead to localised corrosion of waste packages surface and welded areas. In the revised performance assessment abstraction, localised corrosion can only occur if seeping water is able to contact the waste packages. The updated risk insights suggest evaluation of processes that could affect seepage water chemistry may be warranted to better understand the importance to waste isolation.

Conclusion

The preceding discussion provides an example of how independent performance assessment may be used to obtain initial risk insights that can provide a basis for updating and improving the performance assessment information in prelicensing activities. The improved data and models can, in turn, provide improved risk insights. Engaging in this iterative process of developing independent performance assessment capability enables the regulator to better understand how complex processes and their associated uncertainties can influence repository performance estimates. This iterative process helps to improve risk insights during the prelicensing period so that regulatory reviews of an applicant's safety case or a license application can appropriately focus on the processes and uncertainties that are most important to the demonstration of repository safety.

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Appendix D

SESSION IV

THE EMBEDDING OF THE SAFETY CASE IN SOCIETAL DIALOGUE AND DECISION MAKING

THE PARTNERSHIP EXPERIENCE ON THE DISPOSAL OF LOW- AND INTERMEDIATE-LEVEL SHORT-LIVED WASTE IN BELGIUM

P. De Preter, W. Cool, E. Hooft, A. Waffelaert
ONDRAF/NIRAS, Belgium

J. Blommaert, J. Draulans
STORA Dessel, Belgium

Introduction

With the governmental decision of January 16, 1998, the long-term storage option for the low- and intermediate-level short-lived waste (category A waste) was abandoned and ONDRAF/NIRAS was given the mission to further examine the options of deep and surface disposal, in order to prepare a federal decision on the technical option to be developed. ONDRAF/NIRAS was also asked to develop the methods and structures of interaction with the local stakeholders, and to limit its siting activities to nuclear and candidate municipalities. This brought ONDRAF/NIRAS to the concept of local partnerships with interested municipalities, and during the pre-project phase 1998-2006 partnerships were created with the municipalities of Dessel (STOLA-Dessel, 1999), Mol (MONA, 2000) and Fleurus-Farciennes (PaLoFF, 2003).

On 23 June, 2006 the Belgian Government decided that category A waste will be disposed of in a near-surface repository on the territory of the Dessel municipality. This decision implies that ONDRAF/NIRAS, in further interaction with the local stakeholders, can start the preparation of a licence application.

This decision was the endpoint of the pre-project phase (1998-2006) and was based on the final reports of the partnerships of Dessel (STOLA-Dessel, now STORA) and Mol (MONA), approved by their municipality councils, and on ONDRAF/NIRAS' final report, confirming the feasibility of the proposed disposal systems. As the municipality council of Fleurus did not approve the final report of the partnership PaLoFF, ONDRAF/NIRAS did not submit this report to the responsible minister.

The preceding positive local decision in both Dessel (May 2005) and Mol (July 2005), and both on the partnership and municipality council level, to accept, under certain conditions, a disposal facility on their territory was the result of a 6 years long process of discussions within the partnership of all aspects of the disposal system and its integration in the municipality. During these discussions, both near-surface and deep disposal were considered as relevant options. The partnerships of both Dessel and Mol expressed in their final report a "no preference" for one of the two options.

This paper focuses on the STOLA-Dessel experience, but the main observations and conclusion have a broader validity. We try to identify the main factors that led to local stakeholders having confidence in the safety and feasibility of the developed disposal systems, as expressed in their decisions to accept, under certain conditions, the disposal of category A waste in their municipality.

The decision making process

The governmental decision of January 1998 gave ONDRAF/NIRAS the mandate to start a dialogue with interested municipalities to discuss the acceptability of disposal of category A waste on the municipality territory. Finding out if and under what conditions (technical, social, economical and political) a local acceptance of category A waste disposal exists was the main objective of the pre-project phase. The initial idea was a three years phase (1999-2001), but it turned out that more time was required on the local level (1999-mid 2005).

After the national decision of 1998, and before the national decision of 2006 was taken, the focus and weight of the decisional process completely shifted to the local level. The concept and functioning of the partnerships was the subject of a NEA Forum on Stakeholders Confidence workshop [1]. As has been emphasised in many documents and presentations, the importance of a well-defined, fair and broadly accepted decisional process is crucial for reaching situations of broad acceptance of negatively perceived projects like radioactive waste disposal. Some of the characteristics of Belgian decisional process, and certainly the "local period 1999-2005", have created a situation where a local decision of conditional acceptance became possible.

- With the national decision of 1998 the programme could enter a phase with a well-defined objective and scope, i.e. to prepare the elements to enable the federal government to make a choice between surface and deep disposal of category A waste, by limiting the siting activities to nuclear and candidate municipalities.

  Although giving a clear scope to the programme, it also created frustrations at the local level, because the really difficult issue – disposal of the high-level waste – was out of the scope of the partnerships.

- After the strategic "national" decision of 1998, the decisional weight shifted totally to the local level, and the empowerment of the local level was instrumented with a veto right; each partnership could unilaterally decide to withdraw from the process of dialogue with ONDRAF/NIRAS.

  However, during the discussions in the partnerships the lack of a clear link between the local decision and a consecutive national decision (i.e. what became the 2006 government decision) created the uncertainty at the local level if and to what extent the federal government would respect the local decision.

- The partnerships were created by the municipality council and ONDRAF/NIRAS, giving them a firm local political basis.
- The concept of the partnerships as an instrument of direct democratic participation in decision making was developed by university experts in sociology (University of Antwerp, University of Luxembourg) and the first preparatory discussions with the local decision makers on the possible creation of a partnership were conducted by the university experts.
- The decision of local acceptance was from the beginning (1999) seen at two distinct levels, the partnership itself and the municipality council. Both levels had to be positive before a signal of acceptance could be transmitted to the national level. This created some degree of political neutrality or independence for the partnership; also, a confirmation of the partnership decision by the municipality council gave the required political weight to the local decision.

- The partnerships were created and structured as a non-profit organisation, with a firm legal basis and internal rules of decision making.

- The main local groups (political, economical, cultural, social, …) were represented in the partnership. Each of these representatives had a relatively broad network in the local community.
- Every local citizen could become an independent member of the partnership; he could decide to what themes of discussion he wanted to contribute (siting and design, safety, environmental and health matters, local integration). The possibility of direct participation in a decision process was for some of the independent members an important reason to step in.
- The themes and issues discussed in the four working groups (see Section 3) were chosen to a large extent by the working group members.
- Each partnership produced its own final report. After approval by the municipality council ONDRAF/NIRAS submitted this report to the responsible minister.
- The partnerships made considerable efforts to reach out for the wider community by networking of its members and by a wide variety of interaction and communication initiatives (including polls and open door days).
- Flexibility of timing was allowed to come to an informed decision on the local level, leading to an extension of the partnership lifetime from 2 to 6 years.

Process and organisation of information exchange, knowledge building and final reporting

Each partnership was faced with the question if and under what conditions a disposal facility of category A waste on its territory would be acceptable. The three sub-questions structuring the work were the questions of safety and protection, of feasibility (technical and siting feasibility) and of local integration of the project.

The central entities of technical discussion within the partnerships were the three technical working groups “Siting and design”, “Safety” and “Environmental and health matters”. A fourth working group (“local development”) discussed the issues of local integration of the repository. Each working group had typically 10 to 15 local members and one ONDRAF/NIRAS representative, and met once every one or two months. The most active working group (siting and design) met 40 times in the period 2000-2004.

In order to correctly deal with the challenges of knowledge build up, of discussion and integration of information, and of a reporting that correctly reflects the discussions and elements of consensus, each of the working groups went at its own pace through three periods of functioning.

Information acquirement period

There is clearly a tension between a broad participation process and the complexity and multidisciplinary nature of the subject of radioactive waste disposal (mass of information, different expert fields, the challenge of the unusually long time scales and difficult to grasp physical and biological entities, such as activity and dose). Most members of the working groups had no or little familiarity with the issue of radioactive waste disposal.

This created a need for a rather long period of information acquirement through presentations by ONDRAF/NIRAS staff or external experts, through technical visits or participation at workshops, symposia and conferences. In this way the heterogeneous group in terms of professional background, familiarity with the subject evolved into a more homogeneously informed group. The members of the groups largely determined the themes requiring specific attention during this period.

This initial period was also needed to better define the boundaries between the three technical working groups and the scope of work of each group. Nevertheless, this issue remained a concern throughout the lifetime of the partnership. The working group on “siting and design” quickly became the leading and most dynamic working group, with a clear objective and working plan, while the other two technical working groups continued to struggle with the definition of their respective role, scope and plan of work. This was to a large extent a consequence of the more concrete nature of the topics discussed in the “siting and design” working group, amplified by the decision of the working group to involve external experts in the assessment of the information that ONDRAF/NIRAS provided. Also, the fact that the two “evaluation” groups (“safety” and “environmental and health matters”) had to wait till the “siting and design” working group produced its first output was contributing to the lack of clarity in this first phase.

Study and evaluation period

This period took most of the pre-project phase. The various elements of importance for each working group were discussed.

In the “siting and design” working group the discussions were structured around the main components of the repository, as well as the main phases of repository development (construction, operation, closure, institutional control), leading to a well structured and organised approach.

In the two evaluation groups less structuring elements were available, and the subjects of discussion were most often decided on an ad hoc basis. In the “safety” group the main elements of the long-term safety assessments provided some structure (general safety approach for disposal, safety criteria, characteristics of the waste that are of importance for long-term safety, FEPs (features, events and processes important for the safety of the system) and scenarios, results of assessment calculations, treatment of specific scenarios of interest, e.g. airplane crash).

At the end of this period the main results of the “siting and design” group were presented to the “safety” and “environmental and health matters” groups for discussion and assessment. As the two evaluation groups had not been involved in the process of the detailed discussions of the “siting and design” group, they were not really able to have an in depth assessment and judgement of this groups outcome (i.e. a reference design and a site).

Conclusive discussions and reporting period

At the end of the pre-project phase a long time was taken (approximately 1 year) within each of the working groups to report on the work conducted. Here also the “siting and design” working group took the lead, and its final working group report was the central report around which the other technical working groups formulated their results.

The information integration challenge of the partnership

Throughout the three periods of work described in Section 3, a major challenge for the partnership was to integrate all the information in a broadly shared global view.

The initial idea was that this integration would be achieved in three ways:

1. direct interactions between the working groups, e.g. through joint meetings or through regular contacts between or meetings of the presidents of the working groups;

2. the discussions of the results of the individual working groups in the partnership body supervising the working groups, i.e. the partnership council;
3. report and redaction activities by the two persons permanent staff of the partnership, with one person with a technical profile and one with social profile).

The first two mechanisms turned out to be rather weak integrators. With respect to the first mechanism, the "siting and design" working group took also up in its discussions the safety considerations (operational and long-term) related to siting and design when discussing and deciding on repository components and on the way to construct, operate, close and monitor the facility. In fact, because of the role and mission of this working group it was almost forced to take into account in a qualitative manner safety issues. With regard to the second mechanism, the partnership council heavily relied upon the working groups for the technical discussions and limited its role to non-technical matters.

As a consequence, the technical permanent staff member of the partnership became a key integrator of information, especially in the third period of reporting.

Confidence in the safety of the repository

The positive decisions, on the level of both the partnership and the municipality council, were an expression of confidence in the safety and feasibility of a potential repository for category A waste in the municipality. The experience shows that this confidence can be attributed to a few basic elements, the decision process (Section 5.1), the quality of the information provided (Section 5.2) and of the system discussed (Section 5.3), as well as controls of the disposal system and information and knowledge preservation (Section 5.4).

The process of decision preparation and decision making

The partnership experience is an illustration of the importance of the process of decision making for gaining broad local acceptance. Some crucial elements are listed below:

- Everybody willing to participate was be able to do so; all inhabitants of the municipality were invited to participate, and could step in, even during the process.
- The information that was needed to take an informed decision was easily accessible; all members could ask for presentations on a specific topic. For most members the information made available during the meetings in the form of presentations turned out to be more important than the information contained in reports. Reports, however, constituted the most important source of information for the external experts called upon by the partnerships.
- Participants in the process could have a certain influence on the decisions, but also had to respect and accept group decisions. The impact on decisions was possible at different levels, e.g. going from election of working group presidents, to the invitation of external experts, definition of topics for the meetings, up to the final decision of conditional acceptance.
- Participants could observe an openness and willingness from the side of ONDRAF/NIRAS to take into account suggestions, concerns, criticism, …
- The boundary conditions or scope of the process (e.g. discussion of category A waste only) were clearly defined, but as mentioned in Section 2, this point was heavily challenged by many members of the partnership, and ended up in the requirement formulated in the final partnership report that future interactions should be broadened to all radioactive waste issues.

- The presence in the decision process of critical persons is important; they rouse discussion and critical thinking. External experts, including the regulators, that scrutiny the programme can played an important role in this respect. However, criticism with the sole objective of obstruction was promptly auto-corrected by a group motivated to work in a constructive manner.

The quality of the information provided

In the pre-project phase of the programme the qualities of the developed solution were mainly judged on the basis of the quality of the information provided. Through a judgement of this quality or lack of quality, the capacities of ONDRAF/NIRAS as a future implementer of the disposal system were also judged:

- The quality of the information provided, as well as the way it is provided (defensive?, arrogant?, open?, humble?, professional?, enthusiastic?, unmotivated?, unconnected?, well prepared?,…) was the first way for the partnership to judge the quality of ONDRAF/NIRAS as an organisation.
- The information presented has to be coherent and non-contradicting. Different persons from ONDRAF/NIRAS have to bring a coherent message and respond in a coherent way to questions. The work and results presented by contractors to ONDRAF/NIRAS have to fit in the main messages and the general approach of ONDRAF/NIRAS.
- The information provided could withstand scrutiny by the working groups and by external experts. Also, ONDRAF/NIRAS was seen as an organisation willing to take into account comments, corrections and suggestions. There was a general feeling that no negative information was kept secret.
- People often judge on the basis often a general "does it make sense?" impression (common sense feeling). The partnerships expressed e.g. a marked preference for simple and robust solutions, while a high-tech monitoring system with questionable reliability was negatively perceived.
- Tailored, well-dosed (one theme or topic) presentational information packages are a pre-requisite, while too academic and complex presentations lead to frustration. For each presentation the audience has to understand where the treated topic fits in the overall picture.

In depth discussions of the main elements of the system

While the quality of the information provided is a general requirement, the intrinsic quality of the discussed disposal system was made apparent during the discussion of specific topics in the working groups.

- The characteristics of the site contributing to safety and the weaker elements of the site that require reinforcement of some elements of the design of the facility.
- The characteristics of the waste (radiological and chemical) and the understanding that through radioactive decay over a few centuries the short-lived fraction disappears, while for the remaining long-lived fraction assessments have to show a continued safe situation.
- The role of each component of the system and the way it contributes to safety; this qualitative discussion was definitely more prominent and more influential for confidence building in the working groups than quantitative assessments of dose and risk.

- In case of surface disposal, and because of its location within the biosphere, the possibilities to check and monitor the system or parts of the system over a long period of time. Retrievability was seen as a logic consequence of a strategy to control the functioning of the system. For deep disposal in the Boom Clay the limits of monitoring and retrievability were clearly acknowledged, and were considered to be less required, mainly because of the high performance and robust nature of the natural clay barrier.
- The timescales over which the system should operate, with the short-lived activity disappearing at the end of the institutional control period, but with the long-lived fraction and the stable chemo-toxic elements still present.

The working group members critically and extensively discussed the more concrete issues, such as the various barriers or components of the facility, the organisation of waste transport and handling, and repository operation, or the potential mechanisms of component failure. More abstract and more difficult to grasp ideas and calculations, such as dose calculations, were not really discussed. In the safety assessment part most attention went to the discussion of FEPs and scenarios, because of its more concrete character.

Also, the discussions in the working groups illustrated that people often think in terms of comparisons:

- is the category A level waste really a problem compared to the high-level waste?;
- is it possible that components will remain intact over extended time periods when we see in our daily live examples of degradation of components like e.g. concrete;
- is the radiological content of the category A waste not negligible compared to the non-decaying chemical content ?
- how can we argue in terms of zero risk if all what we do entails some risk? the same holds true for the discussion of zero releases from a system over thousands of years.

Additional elements of confidence: controls and knowledge transfer

There is a general understanding and acceptance that disposal systems are designed and developed to become passive systems. Its long-term safety should not depend on future actions, such as surveillance and maintenance. The partnerships, however, put a clear emphasis on specific actions of extended monitoring (in the case of surface disposal). The transfer of information and knowledge to future generations and the efforts to maintain the expertise were seen as additional and complementary lines of defence.

The maintenance of scientific expertise and knowledge, on a local level, to judge the safety of the disposal system was formulated as a crucial condition for local acceptance.

The role of a safety case in the decisional process

The partnership experience, with the positive decisions on the local and national level for surface disposal of the category A waste in Dessel, indicates that the elements and arguments leading to confidence in and acceptance of a disposal solution on a local level are very broad, and certainly much broader than the claim or statement that calculated impacts remain under the regulatory safety criteria. The scrutiny and assessment of the disposal system in the partnership was in the first place on the level of the roles and functions of all the system components, and on the possible failures and degradation mechanisms leading to poor or insufficient performances, as based on expert judgement.

Most of the elements that, together, constitute a safety case (safety strategy, system concept, system understanding, assessment tools, and discussion of evidence, analyses and arguments) were discussed to some extent in the partnerships. The way these elements were discussed was mainly determined by the organisational structure of the partnership (working groups), and not by the report structure of a planned safety case. The focus was on the most concrete elements of the system (site, design, construction and operation).

The discussions on the design of the facility in the "siting and design" working group already integrated in a qualitative manner the main considerations of operational and long-term safety (simplicity and robustness of the system, main safety function and durability of components,…). These discussions had a major impact on the decision of acceptance of the developed solution. Compared to this, the discussion of safety assessments and the results of the assessments (dose calculations) contributed less to the broad confidence, but they do constitute the required confirmation for this confidence.

The documentation structure of the partnerships final report reflected the structure of the working groups and the themes discussed in the working group meetings. This documentation structure was also clearly linked to the federal decision to be taken: which of the developed disposal projects would the government choice to bring to the phase of licence application. So, the final report contained information on:

- the site selected by the partnership and the reasons, technical and other, why this site was selected;
- the proposed design of the facility resulting from the in depth discussions of ONDRAF/NIRAS initial generic reference design;

as well as the argued statement that the partnership as a general, and the working groups "safety" and "environmental and health matters " in particular, were convinced that the disposal facility can ensure safety and protection.

References

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NYE COUNTY NEVADA LOCAL PERSPECTIVE OF THE YUCCA MOUNTAIN PROJECT (YMP)

D. Swanson
Nye County Nuclear Waste Repository Project Office, USA

Abstract

Nye County, Nevada, is host of the proposed Yucca Mountain nuclear waste repository. The Department of Energy's (DOE) Nevada Test Site (NTS) and the Department of Defense's Nevada Test and Training Range occupy a large portion of Nye County. The NTS has been the site of numerous nuclear device detonations; hosts two low-level nuclear waste landfills; and was (and is) the site of various nuclear physics experiments and tests that have resulted in the distribution of radionuclides into the environment.

The Nuclear Waste Policy Act Amendments of 1987 designated Yucca Mountain as the only site, of the three sites approved for characterisation, to be evaluated as a repository. The Act includes provisions for local involvement in program oversight. Nye County and each county surrounding Nye is designated an "affected unit of local government" (AULG). Nye, being the situs county, also is provided the opportunity to have an "on-site representative". This function is a day-to-day opportunity to interact with DOE staff and be actively involved in the DOE decision-making process.

DOE has recognised Nye County's unique status and special needs and has provided additional funding for various studies via co-operative agreements. The most notable program is the County's Independent Science Investigation Program (ISIP). This unique program allows Nye County to contract with subject matter experts, primarily hydrological and geotechnical experts, to conduct studies and advise the county regarding their results and the technical results of DOE's investigations. Through the ISIP, Nye has developed a co-operative and credible relationship with numerous research facilities including the national laboratories, government agencies, and universities.

Nye County has no viable means to reject the YMP. Hence, current County policy is of a pragmatic nature in that our objectives are to assure that public health, safety and the environment are adequately protected, that the YMP is a success in every way possible, and that Nye County benefits economically from the project.

Introduction

Nye County, Nevada, is host of the proposed Yucca Mountain nuclear waste repository. The county is rural with a population of 44 500 residents, most of who reside in the southern portion of the county. With a land area of 11 million acres (4.4 million hectares), Nye is one of the largest counties in the United States. However, 98% of the county's land is publicly controlled leaving only 2% of the land to contribute to the tax base.

The Department of Energy's (DOE) Nevada Test Site (NTS) and the Department of Defense's Nevada Test and Training Range occupy a large portion of Nye County. The Yucca Mountain site is partially contained within the NTS. The NTS has been the site of 1 000 nuclear device detonations, over 800 of which have been underground tests; hosts two low-level nuclear waste landfills; and was (and is) the site of various nuclear physics experiments and tests that have resulted in the distribution of radionuclides into the environment. Nye County residents have had a long association with things nuclear.

Nye County plays a unique role in the Yucca Mountain repository program. As the situs jurisdiction that will forever host the Yucca Mountain repository if it is licensed and constructed, Nye exercises oversight authority specifically granted by Congress under Sec. 116 of the Nuclear Waste Policy Act, (NWPA) as amended. The NWPA, first enacted by Congress in 1982, authorised the Yucca Mountain program. Under that Act the DOE was charged with the responsibility of selecting three sites for characterisation, and from those three choosing one for development as a repository. In 1987 Congress amended the NWPA to focus the entire effort on Yucca Mountain alone, and prohibited further consideration of any other site.

The NWPA refers indirectly to Nye County, for it defines "affected unit of local government" as "the unit of local government with jurisdiction over the site of a repository or a monitored retrievable storage facility." Nye County is that unit of local government. The term may also, at the discretion of the Secretary of Energy, mean local governments that are contiguous to Nye, and the remaining 7 counties in Nevada and Inyo County, California, which comprise the Affected Units of Local Government (AULG), derive their authority from that sentence. Only Nye is specifically called out by definition in that Act, however, and thus only Nye is a "situs" Jurisdiction. Additionally, only Nye and the State of Nevada are entitled under the Act to an on-site representative. Section 117(d) of the Act requires the Secretary of Energy to offer to the State or "any unit of local government within whose jurisdiction a site for a repository or monitored retrievable storage facility is located under this title an opportunity to designate a representative to conduct on-site oversight activities at such site." The County has had an On-Site Representative since 1992, with an office located in the DOE Yucca Mountain Project offices in Las Vegas, Nevada.

Section 116 of the NWPA governs Nye County's oversight authority. Subsection 116 (c)(1)(B) requires the Secretary to make grants to the State, Nye, and other AULG to enable them:

(i) to review activities taken under this subtitle with respect to the Yucca Mountain site for purposes of determining any potential economic, social, public health and safety, and environmental impacts of a repository on such State, or affected unit of local government and its residents;

(ii) to develop a request for impact assistance;

(iii) to engage in any monitoring, testing, or evaluation activities with respect to site characterisation programs with regard to such site;

(iv) to provide information to Nevada residents regarding any activities of such State, the Secretary, or the Commission with respect to such site; and

(v) to request information from, and make comments and recommendations to, the Secretary regarding any activities taken under this subtitle with respect to such site."

In the aftermath of site approval, DOE has taken the position that site characterisation activities have ceased, and the department is no longer willing to fund monitoring, testing or evaluation activities under subparagraph (iii). Nye County's oversight program is thus now focused on activities authorised by paragraphs (i), (ii), (iv) & (v).

Protocols and Agreements with DOE

Supplementary to and in the execution of its oversight authority under the NWPA Nye has entered in to series of agreements and protocols which further define the County's relationship with DOE and the Yucca Mountain Program. These include the *Framework For Formal Interactions Between Nye, County Nevada and the U.S. Department of Energy/Office of Civilian Radioactive Waste Management*, executed in 1991, the *Protocol Addressing Procedures For Nye County On-Site Representation During Yucca Mountain Project Site Characterisation Activities,* executed in 1992, and *Access And Procedures For On-Site Independent Verification And Testing*, executed in July of 1994 as *Appendix A* to the *On-Site Representation Protocol*, which forms the cornerstone for the County's Independent Scientific Investigation Program (ISIP).

Aside from the protocols listed above Nye has also entered into a series of co-operative agreements with DOE, and received funding (similar to a grant) pursuant to those agreements (independent of and in addition to the oversight funds provided directly by Congress) for Nye's independent technical programs. These technical programs, or special projects, have included a significant Independent Scientific Investigation Program (including an early warning drilling program), a Transportation impact evaluation, and a public safety program. Each of these programs is intimately integrated with similar programs of the DOE, but focused on specific concerns of Nye County.

Oversight and On-site Representative Program

Even though Nye County has no choice to accept or reject the Yucca Mountain site, the county's government and leadership have done what it could to protect its residents, and its economic future since 1983. In carrying out its oversight role under the NWPA, Nye has always been guided by these underlying principals: assurance of the health and safety of its residents now and in the future, equitable treatment in transportation, and a viable, attractive economic future.

In its oversight role, Nye County reviews and comments on DOE documents, including the environmental impact statement and site recommendation documents that present the DOE's case that public safety and the environment will be adequately protected. The county also is represented at meetings held by oversight bodies, including the Nuclear Regulatory Commission (NRC) and the Nuclear Waste Technical Review Board, to monitor the discussions and present the county's views when appropriate. The county also comments on the safety standards and licensing requirements proposed by the Environmental Protection Agency and NRC, respectively, as these requirements will become the basis for decision making on continued repository development. The county will review the DOE's license application and supporting documents once they are submitted to the NRC for authorisation to construct the repository. The county will also participate in the formal licensing proceeding.

The Nye County oversight program began in 1987 when the county established the Nuclear Waste Repository Project Office (NWRPO). The county's primary concern was whether a high-level nuclear waste repository at Yucca Mountain, or an interim waste storage facility at the Nevada Test Site would pose unacceptable risks and impacts to the health, safety, and well being of Nye County residents. The county began receiving federal funding in 1992, but annual funding since that time has been unpredictable. Due to the unpredictability of funds, the NWRPO operates with a small permanent staff and a large contingent of consultants who have unique experience and abilities.

In line with the authority provided by the NWPA, the oversight and on-site representative program has strived to meet the following goals:

1. Develop and maintain the capability to monitor and access the health and safety aspects of past, current and potential future activities on federal facilities within Nye County.
2. Encourage a co-ordinated Nye County involvement in all appropriate aspects of the planning, development, operation, and monitoring of the Yucca Mountain waste repository.
3. Maximise the economic benefits of existing, new and reconfigured federal facilities located within Nye County by increasing the procurement of goods and services locally, encouraging workers to live in Nye County, and encouraging the relocation of businesses to Nye County.
4. Encourage federal facilities to fully integrate into local community development initiatives.
5. Monitor evolving federal policy regarding the YMP, and ensure that Nye County citizens are informed of policy decisions and their effects.
6. Provide Nye County with the opportunity to provide pre-decisional input on all aspects of future DOE activity in Nye County.
7. Provide DOE and the federal government with input from the community's perspective.
8. Provide the resources required to craft Nye County responses to contingencies that will arise during or subsequent to YMP implementation.
9. Seek reliable funding to monitor, test, and evaluate environmental, health, transportation, and socioeconomic effects.
10. Co-ordinate data collection efforts with those of the relevant federal and state agencies.

Nye County must be entitled to participate as a full party to the licensing process as part of our oversight role. This has always been envisioned by all parties to this process, and is supported by the language of the NWPA itself. The County has, since 1988, participated in the development of the Licensing Support Network (LSN), and the NWRPO Regulatory & Licensing Advisor serves on the Nuclear Regulatory Commission's (NRC) LSN Advisory Review Panel. Nye has developed and is operating its own LSN web site, where NWRPO licensing documents will be posted and available to all other potential parties. Prior to licensing, these documents will be available through the NRC-managed LSN in accordance with the NRC's requirements in 10 CFR Part 2.

A critical and significant part of the Nye oversight program over the years has been to prepare the County to participate fully in NRC licensing in order to protect the health and safety of current and future Nye County residents, as well as the environment. The County's posture, and the issues it feels are significant and wishes to address in order to best represent the interests of the County and its residents, will thus be decided by the County's leadership at the beginning of the licensing process. Whatever that decision might be, the NWRPO will be prepared to engage in licensing in a constructive and positive manner, and to carry out its essential mission of guarding the health and safety of Nye's residents.

Special Projects

Special Projects typically are highly focused efforts designed to evaluate a particular issue or problem. Goals and objectives are clear and concise. In the past, these projects have dealt with scientific issues of particular interest to Nye County. Thus far, most of these projects have been funded by means of Co-operative Agreements. Each agreement specifies the tasks to be undertaken, the cost of various elements of the tasks, task schedule, and the personnel to be involved. As implied, both Nye County and the DOE co-operatively agree upon the work to be performed.

Once the general outline is agreed upon, however, the contractual document limits the DOE's ability to direct the day-to-day work. For example, a co-operative agreement specifies: "*DOE will be limited in its authority to specify how work will be conducted and will not have the authority to make specific work requests at the task assignment level. Nye budgets and expenditures, except as associated with proposed tasks and annual budget approvals, will be the sole responsibility of the recipient*." DOE's involvement is limited to appropriate technical direction that is described as "substantial". In other words, the DOE will be principally involved in establishing project goals, objectives, schedule and budget, but DOE will not be involved in how these milestones are actually attained.

In accepting an award, Nye County agrees to the terms and conditions of the award. These typically are specified by federal regulation and address such activities as non-discrimination, environmental protection, intellectual property rights, lobbying restrictions and so forth.

The Early Warning Drilling Program and the Independent Scientific Investigation Program (ISIP) are the largest projects funded by co-operative agreements. This scientific program began in 1995 and will continue indefinitely as long as funding is made available. The scientific and engineering goals emphasise the hydrogeology of the area surrounding Yucca Mountain and down gradient of the mountain. Specific objectives of the ISIP include the following:

1. Provide a better definition of the potential risk to Nye County residents' drinking water supplies from high-level nuclear waste handling and disposal at the Yucca Mountain repository.
2. Design an appropriate "early warning" ground-water quality monitoring network between the repository and present and future populations in the community.
3. Evaluate and monitor the impacts of water usage resulting from YMP development and identify unacceptable consequences of such use.
4. Interact with the scientific and engineering community to better understand, predict, and manage the risks associated with high-level nuclear waste handling and disposal at the Yucca Mountain repository.
5. Provide engineering design suggestions for the Yucca Mountain repository that may reduce risks to Nye County residents while controlling costs.
6. Disseminate scientific information to citizens and other agencies.
7. Strive for the highest scientific standards and challenge other investigators to achieve high standards.

Nye County considers it imperative that this scientific and technical program be conducted independently to maintain an unbiased credibility.

All Nye County technical and scientific work is carried out under a Quality Assurance (QA) Program designed to meet the criteria of the NRC regulation 10 CFR 50, Appendix B, as well as relevant requirements of the American National Standards Institute/American Society of Mechanical Engineers Nuclear Quality Assurance Standard #1 (ANSI/ASME-NQA-1). The Nye QA Program has been accepted by the NRC (NRC cannot "approve" the program since Nye will never be an applicant for a license to operate a repository) pursuant to an Acceptance Evaluation dated March 19, 1999.

A more recent and evolving program funded by a co-operative agreement is the Public Safety and Related Services program. The impetus to initiate this program was recognition that the construction and operation of the Yucca Mountain repository will require substantial emergency response capability on-site. Nye County also anticipates a need for enhanced emergency response capability in the adjacent community as it grows with the project. Hence, the program began as an evaluation of how the county might enhance its existing capability to meet both the repository requirements and community requirements. The conclusion at this time is that both Nye County and the DOE would jointly benefit programmatically and economically from an expanded county capability.

A special project initially funded by a co-operative agreement and subsequently funded with oversight funds relates to evaluation of the positive and negative impacts of the railroad that will be constructed in Nevada beginning at existing tracks and extending to the repository. Transportation of nuclear waste in Nye County has been a concern since project inception. The principal safety concern is not necessarily that radioactive materials will be transported near citizens. The concern is related to the possibility of increased truck traffic on rural highways and the potential for more numerous accidents. Hence, an early request to the DOE was to construct rail as soon as possible to facilitate initial repository construction and for subsequent waste transportation because rail transportation was judged to be safer than truck transportation.

A second concern related to rail transportation was the potential economic benefits that might be realised assuming the railroad would be available for shared use in addition to waste transportation. Surveys of existing commercial enterprises in Nye County showed that many firms, especially local mines, would use rail if it were available to them.

Summary

The YMP oversight program managed by the Nye County NWRPO is unique in many respects. The historic success of the program is directly related to the ability of the office to have On-Site Representative status and day-to-day direct access to DOE managers and policy makers. In addition, the office is able to hire, through a variety of funding sources, consultants that provide a very broad expertise capable of dealing with complex engineering and geotechnical issues as well as political and policy issues.

It is important, in the author's opinion, for Nye County to have a broad, independent, capability of assuring the health and safety of its current and future residents potentially affected by DOE activities to transport, store, and dispose of highly radioactive wastes at Yucca Mountain and to provide this assurance, to the maximum extent feasible, consistent with the authority of the local government, rather than by exclusive reliance on the federal government or its contractors. This objective is accomplished by Nye County's statutorily dictated ability to oversee the DOE's work and to conduct independent activities to evaluate the impact, or potential impact, of that work and confirm the adequacy of the DOE's data, analyses, models, and designs.

It is Nye County's opinion that the County must participate in oversight of the program, including the DOE's performance confirmation program, for the life of the storage facility. Nye County oversight, including on site representation, should extend to all DOE activities (risk assessment, engineering design, transportation, waste handling, storage, emplacement and monitoring) including any post closure monitoring, associated with its proposals at Yucca Mountain, and to the cumulative effects of other DOE activity (past and ongoing) in Nye County. These efforts will contribute to the YMP being a "success".

THE EMBEDDING OF THE SAFETY CASE IN SOCIETAL DIALOGUE AND DECISION MAKING

K. Nilsson
Municipality of Oskarshamn, Sweden

Abstract

Oskarshamn is one of two Swedish municipalities where site investigations are conducted by SKB as possible sites for a final repository for spent nuclear fuel. The municipality has been an active part in the process since 1994 when Oskarshamn was pointed out as a suitable site for an encapsulation plant. A project organisation LKO – Local Competence Building was set up 1994 and it is financed by the Swedish nuclear fund.

There are two leading themes that form the basis for our participation – complete openness of plans and results and participation with the possibility to influence.

Site investigations for a repository started 2002 and will be finished when SKB has selected a site, which is planned late 2009. To follow up the site investigations an organisation based on working groups has been set up. The groups are the Safety Group, The Misterhult Group and the Future Perspective Group. About 45 persons are participating in the groups. The members are politicians, civil servants, people representing various associations and interested citizens.

The main goal for the LKO organisation is to present a comprehensive report to the municipality council if SKB will apply for a final repository in Oskarshamn.

When the municipality council accepted site investigations the decision was taken with 13 conditions. The most important conditions are about long term safety.

General Background

The municipality of Oskarshamn is located on the east coast of Sweden, the municipality covers an area of about 1 000 km2 and has about 26 000 inhabitants. Oskarshamn has a stable economy and high employment. Located in the municipality are three nuclear reactors, and the CLAB an interim storage for all spent nuclear fuel in Sweden. We also have the Äspö underground laboratory operated by the Swedish Nuclear Fuel and Waste Management Company (SKB), which is a kind of dress rehearsal for a final repository. Another SKB operated facility in the municipality is a laboratory for development of methods for closure and testing of spent fuel disposal canisters.

In 1992 Oskarshamn was pointed out as the preferred place for siting an encapsulation plant. Between 1997 and 2000 a feasibility study for a final repository was conducted. In late 2000 the municipality received an invitation from SKB to participate in a site investigation programme. In March 2002 a unanimous council voted for participation.

Recently, November 2006, SKB has applied for building an encapsulation plant in Oskarshamn.

The Municipality of Oskarshamn is one of two Swedish municipalities currently subject to site investigations for a possible final repository for spent nuclear fuel. The site investigations follow feasibility studies in eight Swedish municipalities. The implementer, SKB, has plans to file a license application for a repository in 2009. The application must, among other documentation, include a comprehensive safety analysis report, a detailed site-specific systems description and an Environmental Impact Assessment (EIA).

Licensing of a repository in Sweden is subject to two major acts – the Act on Nuclear Activities and the Environmental Code. The Swedish Nuclear Inspectorate (SKI) prepares the government decision according to the Act on Nuclear Activities and the Environmental Court the government decision according to the Environmental Code. The municipality has the possibility to veto a repository according to the Environmental Code.

Introduction

Is a final repository safe? That is the key question for the public and the decision makers in Oskarshamn. Before other aspects of a possible final disposal can be discussed there must be a convincing answer about the safety.

The main difference between a "conventional" project and a final repository is the extreme time span during which the spent fuel will have a high potential to threat human health and environment. The public is well aware and concerned by this fact.

The public is also well aware that the problem must be solved and that it should not be postponed and be a burden for future generations. From several polls there is a clear advice to the experts to continue to develop and test new technology such as transmutation in parallel to the continuous study of geological disposal. In summary the message from the public concerning our spent fuel is – OK go ahead and carry out site investigations for a geological repository but continue to invest in alternative technology where the time span can be shortened.

The consent of the public to a repository will require unbroken trust in the safety case. This does not necessary mean a detailed understanding of performance assessment and all the long-term processes and events that are evaluated. In a complex safety case like this I would argue that there are probably not a single expert, who has a detailed and complete understanding of the entire safety case. In the scientific world various facts, theories and calculation cases are built up to an entire safety assessment. Where lack of knowledge exist and other uncertainties cannot be removed the use of conservative assumptions are needed. Conservative scenarios and calculation cases may also be needed to test the robustness of the system. One can say that if a conclusion is supported by a broad majority of experts it has a tendency to become accepted as a fact and remain to be perceived as a fact until replaced by another conclusion or theory. The public is well aware of this process. There are many concrete examples of this in the evolution of science in the past.

Siting of a repository is not solely a decision by experts. It is nor an issue of taking the risk-based decision making from the experts and give it to the public. It is rather the challenge to facilitate risk-based decisions by a sound and balanced participation by all stakeholders in their respective role. From many experts I hear this is not possible, participation requires a certain level of knowledge etc. – maybe like a green card in order to play golf – I think such an expert position is a threat to progress in waste management.

In a democratic society, final decisions are political and taken by laymen in government and in the case of Sweden also by the laymen in the municipality elected council. The other reality is that the elected decision makers cannot take such a decision without public support and the public does not have resources or time to study the detailed safety case. The decision if a repository is safe or not is in its final stages in the hands of laymen.

How then to establish an acceptable safety case with the laymen decision makers?

What is safety for the public? This is not an easy question. It contends at lot of aspects, science to a point you can understand it, trust for the implementer, regulators and also the politicians in the municipality.

This is a question that has concerned those of us working with the issue of siting a final repository for many years. I am quite optimistic that there are ways to reach rationale decisions also in a complicated safety case as the one for repository long-term safety. There are many aspects of this work and it would take to long to cover them all. I will describe four key aspects:

- A clear description of what the project is all about and what the alternatives are.
- A well defined decision-making process and clearly defined roles of the key actors.
- An open process allowing for participation and influence.
- Facts, values and stretching.

A clear description of what the project is all about and what the alternatives are

Spent nuclear fuel and high-level waste is extremely hazardous for hundred thousand years or maybe more. If not handled correctly in the short term and effectively removed from the biosphere in the longer term it poses significant threat to many generations. Temporary solutions in an interim storage may be acceptable for shorter time periods of say hundred years but for longer term a final solution must be found. The solution that most scientists and experts are suggesting is to finally solve the waste problem by disposing it in a geological formation. This is also what is reflected in the Swedish legislation. This is also what many national programmes initially have aimed for and it has been pointed out as the only solution. But strong public opposition has in many cases brought especially the siting projects to a halt.

From our discussions in Oskarshamn we a clear message about what waste management is all about. There need to be an open discussion about the options – wait and see, long term monitored surface storage, transmutation, geological disposal etc., so that the public and decision makers can among themselves with expert support work out the arguments and reach conclusions. One conclusion reached in Oskarshamn supported by 80% of the public and 49 out of 50 council members is that we shall continue to work on the solution by carry out site investigations in preparation of a licence application. A geological disposal seems to be the preferred option right now. I do not think all programmes have this platform.

A well defined decision-making process and clearly defined roles of the key actors

In complex decision making the format is as important as the content. Changing rules, parallel discussions about process and content, vague roles of the participants and a feeling of that the public is being excluded are ingredients that are likely to stop any nuclear waste project.

A system starting with a clear legislation, a stringent safety standard defined in advance, a clearly defined implementer with authority to propose solutions and sites, a strong regulator with a mandate to follow and review the implementers programme – and stop it if necessary. A local veto or at least a strong local role together with well defined decision-making steps are what I would define as necessary ingredients of a sound national nuclear waste programme.

Three organisations have received funding from the nuclear waste fund to participate in the EIA consultations. One of the organisations, MKG, is very active and has own experts and an expert board. They have now focused on alternative methods and want “deep boreholes” more developed. In the working model for Oskarshamn, the NGOs are defined as a resource in the discussions about the siting process.

An open process allowing for participation and influence

Reality is that the public does not have much interest in a national R&D programme for how the nuclear waste is to be handled and disposed. The interest will arise when the finger is put on the map and potentially interesting sites are identified. From an expert point of view the programme has probably been running for decades and solutions are seen as mature. For the public a completely new issue is now on the agenda and they see this as the starting point and also see it as their right to question all and any aspect of the proposal. This is frustrating to the experts as they already see many questions as finally solved. Maybe this is the most critical point for any nuclear waste project and where probably an explanation is to be found why some projects fail.

At this juncture the experts and laymen must meet, allow time for discussions and be prepared to give and take. If the experts takes on the role that they know all the answers and only takes it as an information problem there is again a large likelihood of failure.

The key words are take the time required for participation and influence. How can the project be set up to allow true participation by the local public and the local decision makers and how can the local expertise be utilised to form a better project? The answers to these questions are keys to progress in any siting programme. For those experts and managers that do not think there is anything to learn from the public and local decision makers and that participation is only a burden I can foresee large problems and distrust.

Facts, values and stretching

With this fourth point I will try to address how to formulate the answer to the question from the public – is it safe?

Important is to set up various forums where expert and laymen can meet. From our experience it is of crucial importance that the experts from the regulator participate in these forums. It is also important that experts with other opinions get the possibility to meet the public.

Safety analysis does no only include facts but also contain value judgements on several levels.

My experience from Oskarshamn tells me that the answer to the question – is it safe? – contains two main components namely how the soundness of the system itself is perceived and secondly can the experts be trusted. The experts can roughly put in three groups – those who are developing the technical details (for the implementer) those who are critical and maybe also linked to the environmental groups and those working for the regulator who review and approve. The role of the regulator is often underestimated. A strong, competent regulator with the legal tools and resources to participate is crucial based on our experience in Oskarshamn if we are to reach solid and respected decisions.

For the critical experts from e.g. the environmental groups we must find forums where they can address their questions to both the implementor and the regulator. The RISCOM project has developed a model in which this can be handled and were stretching is one important aspect of establishing authenticity by the experts and trust in the extension in their recommendations (Andersson *et al.* 2004). To establish a demanding environment for the experts where the issues can be thoroughly stretched is one important ingredient if decision makers and the public will trust the safety case or not.

How do we work in Oskarshamn?

A key factor for our involvement is that we receive independent funding for our work related to the waste programme, this year we receive approximately 700 000 EUR. The government changed the rules so the municipality work could be financed by the national nuclear waste fund.

Another key factor is the strong and independent position of a Swedish municipality. For siting a nuclear facility the municipality has a veto possibility. Some other factors are:

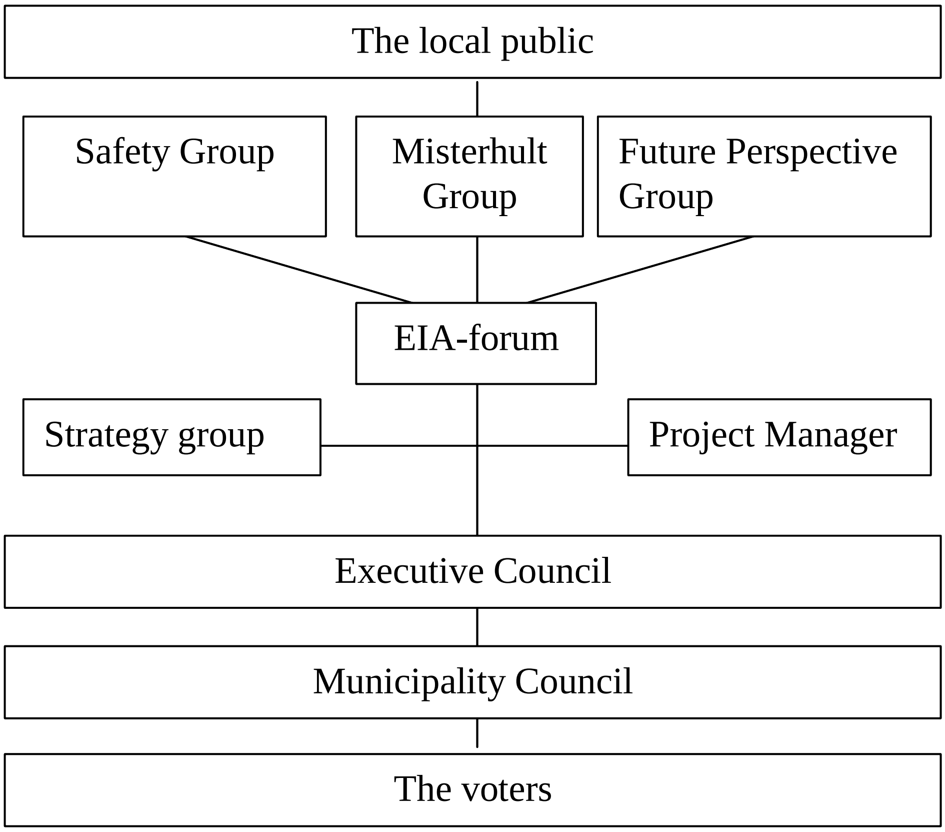
- Local experiences with nuclear operations.
- Traditionally regulators and authorities have a high degree of credibility in Sweden.
- There are clear party roles in the process with the government, the regulators and the industry, SKB.
- A step by step process.

For the municipality there are two leading themes that formed the basis for the participation – complete openness of plans and results and participation with the possibility to influence. These two themes have been formed into seven guiding principles – the Oskarshamn model:

- Full openness, participation and influence.
- The EIA the legal framework.
- The municipality council the local client.
- The public a resource.
- The environmental groups a resource.
- The regulatory authorities our experts.
- Stretching of SKB och the regulators for clear answers.

The municipality formed an independent project directly under the executive council – project LKO, Local Competence Building. The project has continuously been developed but it's general structure has remained since 1994. The current core of the project is three working groups – each one with a number of responsibilities decided by the municipality council. Together there are about 45 members in the groups, politicians, civil servants, landowners, representatives from various associations, representatives of local environmental groups and interested private citizens. The groups meet about once a month, mostly half a day. Each working group has an assigned contact person from SKB who partially attends the working group meetings and provide information on plans and results. The working groups often invites specialists from SKB, SKI the Swedish Nuclear Power Inspectorate, or SSI the Swedish Radiation Protection Authority, to give presentations and answer questions.

Figure 1. **LKO principal organisation chart**



The work is co-ordinated by a Strategy group chaired by the mayor with participation of the three working groups chairs, the project manager and consulting experts for the municipality (see Figure 1).

All members of the Strategy group attend the meetings of the EIA-forum, which is chaired by the Governor of the county administration. The members in the forum are SKB, the Swedish Nuclear Power Inspectorate SKI, the Swedish Radiation Protection Authority SSI, the county administration and municipality with the Strategy group. The EIA-forum meets about three to four times a year. The EIA-forum is mainly set up for questions from the municipality and is a part of the formal EIA process, it is used to reach agreements on what is needed to be included in an application for consturction of an encapsulation plant or a repository.

The council decision to accept site investigations in 2002 contained thirteen conditions:

- Full economical compensation for the municipality participation.
- Only for Swedish spent nuclear fuel.
- A deepened dialogue with the local public on behalf of SKB as well as SKI and SSI concerning criteria and safety analyses.
- An active regulatory follow up of the site investigations and reporting to the municipality.

- SKB must present how the safety analyses, site selection criteria and the site investigations are coupled and relate to each other.
- A systematic compilation by SKI and SSI of research that does not agree with the results or conclusions presented by SKB.
- Access to private land is subject to volunteer agreements with the private landowners.
- A complete site investigation programme including a social science programme.
- A scooping report for the environmental impact assessment (EIA) must be presented to the municipality council for approval.
- The alternative to KBS-3 presented in the EIA must be subject to a broad consultation.
- The issue of long term responsibility for a final repository must be regulated in Swedish law.

So far four of the conditions are fulfilled. Our intension is that all conditions should be fulfilled when/or if SKB will apply for a final repository in our municipality.

Conclusion

The siting process for a final repository in Sweden has been going on for more than a decade. You need a long period to build trust and to communicate with the citizens. It is a vulnerable process, you can lose the trust in a second and it takes years to build it up again.

On the other hand, a long drawn out process can lose interest among both decision makers and the public. A realistic timetable is important the process must be predictable.

We believe that complicated issues like safety assessments can be communicated among laymen if you have a process for it and an idea for how to carry it out. The RISCOM model can help you to succeed.

We also believe in our Oskarshamn model with full openness in all decisions and which gives the public a possibility to participate.

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Appendix E

POSTER SESSION

TO WHAT EXTENT CAN NATURAL ANALOGUES CONTRIBUTE TO THE SAFETY CASE OF HIGH LEVEL WASTE REPOSITORIES IN ROCK SALT?

U. Noseck, W. Brewitz, I. Müller-Lyda
Gesellschaft für Anlagen- und Reaktorsicherheit, Germany

Abstract

The most important key element of any safety case for a radioactive waste repository is the safety assessment. It quantifies potential hazards to the environment from selected scenarios by means of numerical model calculations. A time frame for the assessment of 1 million years is proposed in the course of the revision of the German safety criteria. Due to the growing uncertainties in the safety assessment within the extended geological time frame multiple lines of evidence supporting the modelling results as well as the conclusions to be drawn will be of increasing importance.

One category of supporting arguments can be derived from natural analogues. Their role in the safety case depends to some extent on the time period which they cover. Especially industrial or archaeological analogues can help understanding and underpinning of short-term processes within the repository. However, the main benefit of natural analogues in a safety case is to increase the understanding of long-term processes.

In this paper it is evaluated in what way natural analogues can contribute to the different elements of a safety case. This is illustrated by discussion of their relevance to specific time frames for the example of a generic HLW-repository in rock salt.

Introduction

The key element for estimation of potential hazards to the environment caused by the repository is the long-term safety assessment. Results from calculations performed with numerical models have to be compared to the protection objectives. However, these calculations are subject to different kind of uncertainties which have to be addressed in the safety case. Thus in a safety case the conclusions drawn from the numerical calculations need to be supported by additional arguments.

One type of arguments can be derived from natural analogues, which play an increasingly important role in the safety case for repositories in deep geological formations. The current interest in different countries is reflected by several recent review projects with emphasis on the application of natural analogue study results in performance assessment [1,2,3]. Within the last years the term "Natural Analogue" has got a much wider meaning and includes man-made analogues as well. In this paper we refer to the definition from NANet project, that "Natural Analogues are investigations of natural, anthropogenic, archaeological or industrial systems which have some definable similarity with a radioactive waste repository and its surrounding environment" [1].

The role of natural analogues in the safety case depends amongst others on the time scale to be covered. In the actual German safety criteria for waste disposal no time frame for the long-term safety assessment is given. Currently, revised safety criteria are under development. These criteria will reflect the recommendation of the German AkEnd panel that the isolation potential for the repository system has to be demonstrated for a time frame of 1 million years [4].

In this paper, firstly a classification of natural analogue studies with regard to the time scale they cover is done. Then it is shown how natural analogues studies, which have been performed in the past by different groups in Germany can contribute to the safety case for a repository in rock salt. Also limitations of the applicability of such studies are pointed out. It is not intended to give a comprehensive description of all available analogue studies but to restrict the discussion to selected examples.

Classification of natural analogues

Natural analogues can be classified by the time period addressed in the study. To some extent this determines their use in the safety case. In this work we distinguish between three different timescales. The major contributions of the studies on each time scale as well as major advantages/disadvantages are shortly described in the following and are summarised in Table 1.

- **Industrial analogues**: Such analogues started earliest 150 years ago and result from “disturbances” of the environment caused by input of constructions, materials, contaminants, etc. produced within the Industrial Age. Clear advantages of these analogues are that at least some materials show higher similarity with materials intended to be used in repositories. Furthermore, in many cases initial and boundary conditions are rather well known, e.g. information about the date of constructions or input of contaminants are available and boundary conditions of the systems have not significantly changed during the time frame of the analogue. These analogues can give important information for short-term processes as the healing of an excavation disturbed zone (see below) or time-independent effects as radionuclide speciation in natural environment investigated for example after the Chernobyl fall-out. These analogues are restricted in the way that they cannot contribute to the understanding of the geological development and/or the long-term stability of formations.

- **Archaeological analogues**: Such analogues cover time frames between the past 10 000 and 150 years. They also result from man-made disturbance of the environment with the main difference to industrial analogues that the materials do not stem from the industrial epoch. Major topic of investigation is the behaviour of materials used as engineered barriers, i.e. for waste forms, waste packaging and buffer or backfill, in particular glass, metals and cement [6]. The time scale is important to receive information about long-term corrosion/degradation rates. As already stated for industrial analogues in a number of cases man-made materials show higher similarity to those intended to be used in repositories than natural ones. This is true for cement, where particular attention has been paid to pozzuolanic Roman cements, such as that found at Hadrian’s Wall, and 20th century cements because both contain the calcium-silicate-hydrate compounds that characterise modern Portland cements and, thus, mineralogically are the most similar analogues to the cementitious materials which will be used in an ILW repository [1]. Furthermore, some materials are only man-made and do not have natural equivalents as for example steel.

- **Geological analogues**: Such analogues usually cover time frames of more than 10 000 years and in most cases more than million years. These analogues mostly result from natural changes in the geological environment, e.g. in temperature conditions, in geochemical and/or hydrogeological conditions causing mobilisation or immobilisation of chemical elements, or in geological conditions leading to isolation or degradation of materials. These analogues can also contribute to the understanding of the behaviour of engineering materials. However, due to the high uncertainties in boundary and initial conditions, its worth lies more in the illustration of the long-term stability of a material, e.g. glass than in the derivation of corrosion or degradation data,. The strength of geological analogues lies in the demonstration of the integrity/long-term stability of formations, the understanding of development of long-term geochemical conditions and its effect on radionuclide migration and immobilisation. These topics are not covered by industrial and only to a limited extent by archaeological analogues.

Table 1. **Classification of natural analogues due to the spanned time scale**

Time frame	Notation	Main topics for application	Remarks
< 150 y	industrial	• short term effects and processes • time independent effects	• initial and boundary conditions rather well known, • high similarity to repository materials • only limited time scale
> 150 y < 10 000 y	archaeological	• behaviour of engineering materials, e.g. container and waste matrix	• initial and boundary conditions known to some extent • certain degree of similarity to repository materials
> 10 000 y	geological	• long-term stability of materials and formations • evolution of hydrogeological and geochemical systems • long-term radionuclide migration / immobilisation	• initial conditions not (well) known, • PA long-term scale covered

In general the direct use of quantitative information from NA studies in long-term safety assessments is limited since it is very difficult to extract hard numerical data from complex natural systems where initial and boundary conditions are usually afflicted by high degree of uncertainty. Most likely industrial and partly archaeological analogues contribute to quantitative information according to the better knowledge of especially initial conditions. However, it was shown in the past that this analogue information is usually not directly transferred into parameters for performance assessments but is used to confirm data ranges or to give upper limit values.

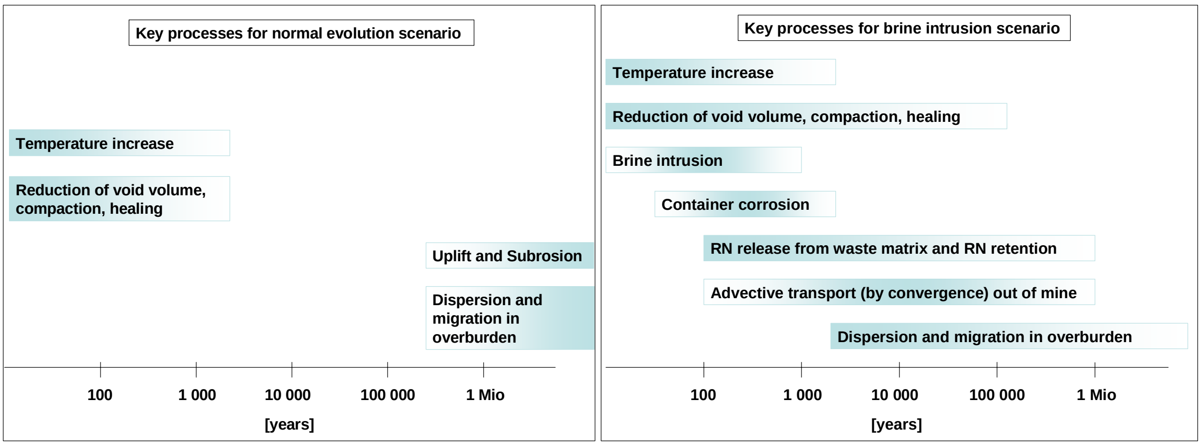
The potential role of natural analogues in a German safety case for a repository in rock salt

For repository concepts for high-level waste in rock salt the geological barrier is of utmost importance, since the normal evolution is expected to lead to a complete and permanent confinement of the radioactive waste in the host rock by convergence of voids. Therefore, beside the fact that more than 250 million years after diagenesis of the evaporates a large number of salt domes exist in the North German Plain, specific arguments from analogues underpinning that the salt dome is stable and integer over geological time frames are crucial for the safety case. In case of an early brine intrusion, which is much less likely, a number of additional processes need to be regarded in the long-term safety assessment. All relevant processes need to be implemented in the models. Natural analogues can also contribute to understanding of these processes and qualification of the respective models.

Figure 1 shows how the key processes for safety assessment can be classified according to their time of occurrence. The normal evolution scenario is on the left. There the key processes can be divided into two groups. At early times processes leading to a reduction of voids, decrease of permeability of seals, and healing of excavation disturbed zones play the key role. They are mainly influenced by the temperature field in the repository area which exhibits significantly increased temperatures over few thousand years. At extremely long time scales rather beyond 1 Million years uplift and subrosion might lay open the area with radioactive waste thereby causing radionuclide transport in the overburden.

In Figure 1 on the right the key processes for a scenario with early brine intrusion are shown. At early times temperature increase, reduction of voids and brine intrusion are the dominating processes in the system. In this scenario back pressure of fluids prolongates the process of reduction of voids by convergence. After access of brine to the emplacement area the corrosion of the container plays an important role and mobilisation of radionuclides from the waste starts after container failure. After the emplacement boreholes and/or drifts have been completely filled with brine, advective radionuclide transport, mainly caused by further compaction of the brine filled voids, occurs. After release out of the salt dome radionuclide migration and dispersion play a role. Highest radionuclide peaks usually occur at early times when the flow rate of the brine out of the salt dome is highest.

Figure 1. **Key processes for the normal evolution and for a brine intrusion scenario for a repository with high level waste in rock salt. The colour of the bars indicates the importance of the process at a distinct point of time: grey = high importance, white = low importance**



Most of the analogue studies worldwide have dealt with non-salinar systems. Nevertheless distinct examples for repositories in rock salt are available. In the following examples for analogues to be used for underpinning of key short-term processes and models as well as analogues to be used for understanding and illustrating long-term processes and integrity of the salt dome are shown.

Analogues for the short-term scale

As an example of an analogue for short-term processes the permeability reduction of the excavation disturbed zone (EDZ) in rock salt observed in an old drift in the Asse mine is described. The excavation of the repository infrastructure with drifts and boreholes changes the favourable properties of the rock salt amongst others by an increase of permeability in a specific zone around the voids. Analogues giving evidence that healing, i.e. decrease of the permeability back to that of the undisturbed rock salt occurs are of great importance for the safety case. Besides its worth for the element "evidence, analyses and arguments" this kind of analogue also contributes to the "Assessment basis" and there especially to the topic "Methods, models and computer codes" by supporting the numerical models describing the time dependent decrease of permeability in the EDZ.

The primary reason for EDZ formation is the stress imposed by excavation that leads to dilatancy which results in an increase in porosity and permeability in a limited zone around the drift wall and the drift floor. Natural undisturbed rock salt is impermeable for fluids because of its low permeability of app. 10^{-21} m^2. The permeability of the excavation disturbed zone can be increased to values of 10^{-15} m^2. Therefore the EDZ might represent a pathway for brine and thereby can facilitate brine inflow as well as radionuclide release from the repository, i.e. geometry, permeability, and temporal development of EDZs are important parameters for performance assessment.

The permeability of EDZs will be reduced when the stress returns to a homogeneous state, e.g. after creep of the rock onto engineered barrier systems as plugs or seals. The models used to describe the reduction of permeability of EDZs with time under high pressure are derived from laboratory experiments. In order to check these models for time frames beyond that of laboratory experiments the analogue site at the Asse was studied.

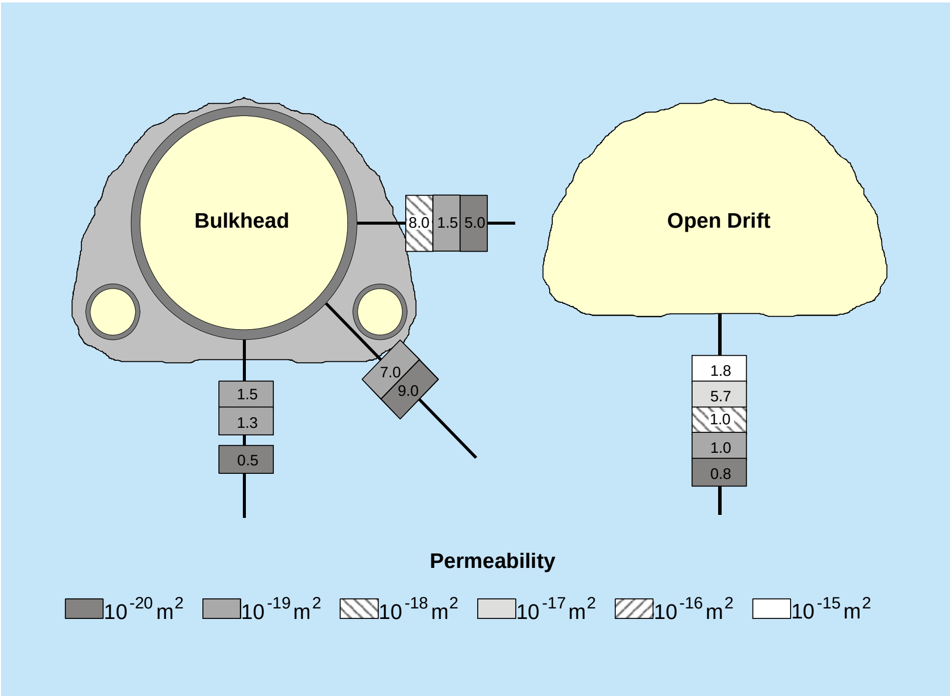
The permeability distribution and the extension of the excavated rock salt were measured at various test sites in the Asse salt mine located in Northern Germany near Braunschweig [7]. One drift, the so-called bulkhead drift, which was mined in year 1911, is especially interesting with regard to potential long-term behaviour of the EDZ. A 25 m long section of the drift was equipped with a liner of cast steel tubings in 1914, and the void between the liner and the drift surface was backfilled with concrete. This drift can be regarded as an industrial analogue for the development of an EDZ in a drift around a sealing. From the stress and permeability measurements taken in the EDZ around this bulkhead, information on the healing process of the EDZ in the long-term can be derived.

Figure 2 shows the permeability of the EDZ around the bulkhead drift compared to an open, unlined drift. Below the open drift, a typical EDZ is present. It extends about 1.5 m into the rock, and the permeability rises above 10^{-16} m^2. This confirms the results of a great number of permeability measurements at other test sites in the Asse salt mine. At all test sites with open drifts, an EDZ extension about 1.5 m into the floor and not more than 0.5 m into the walls was observed. Tests using various setups for measurements close to the open surface yielded permeability increases up to values from 10^{-16} m^2 to 10^{-15} m^2, in comparison to around 10^{-21} m^2 of the undisturbed salt.

Around the lined part of the drift permeability is completely different. Apart from the horizontal borehole close to the drift surface, all permeabilities are less than 10^{-19} m^2 and thus considerably lower than the typical EDZ values. These lower permeabilities are due to the stress state with high normal and negligible deviatoric stress components, which is consistent with the results of supporting calculations. The original permeability of undisturbed salt, however, is not yet attained.

Microstructural investigations on cores from both the lined and the unlined part of the drift seem to indicate that this may be due to the fact that the existing microfractures were closed by stress-induced plastic deformation, but did not completely disappear.

Figure 2. **Measured permeability values in the boreholes around the lined drift (left) and below the open drift (right) in the Asse mine [7].**



Healing of the EDZ is not only a function of stress state, but also strongly time-dependent. The time dependence could not be clearly determined in this study. In case of natural dry rock salt in the Asse mine with about 0.02 wt% water, 90 years under high compressive stress and negligible deviatoric stress were not sufficient to completely heal the EDZ around the bulkhead drift. However, this study clearly shows that a partial healing of the EDZ with permeability reduction of more than three orders of magnitude was observed.

This study shows the worth of an industrial analogue, where the time frame is precisely known. It also shows, that in order to confirm the constitutive models used to describe the EDZ evolution with time more effort is needed. This could not be done by evaluation of only one natural site. Additional in-situ measurements, preferably around excavations of different age and, wherever possible, in the surroundings of plugs placed long times ago, are necessary.

Analogues for the long-term scale

Concerning the integrity of the salt dome on the one hand arguments demonstrating that external fluids from adjacent or overlying strata did not migrate into the inner of the salt dome, i.e. no fluid pathways exist in the formation are of great importance. On the other hand it is important to show that processes like subrosion do not affect the repository within early time scales, when activity of the waste is still high. Analogues exist for both processes. With respect to the elements of the safety case these studies mainly contribute to the section "Evidence, analyses and arguments" since they give evidence for the strength of geological disposal in general and in case of the self-analogue study for the intrinsic quality of the site.

In order to evaluate whether external fluids have been able to migrate into the inner of the salt formation, analyses of gas inclusions have been performed in two different evaporate bodies, in Zielitz and in Braunschweig-Lüneburg [8,9]. Zielitz represents a salt formation which was not much stressed by tectonic activity in its geological history, whereas the Braunschweig-Lüneburg formation was extremely exposed to tectonic load. Gases from grain boundaries and fluid inclusions were analysed.

The gas mixtures in the Zechstein 2 series from Zielitz consist of the components CH_4, H_2, N_2 and H_2S. The component ratios as well as the δ^2H values of CH_4 indicated that fermentation represents the major generation mechanism of methane. The profile of $\delta^{13}C$ in CH_4 is characterised by values of -45 to -50 ‰ (VPDB) in the lower part of the salt sequence and unusually high values of about +21 ‰ in the potassium salts in the upper part of the salt sequence. This profile can be explained by a model which assumes an evaporation basin, which is a closed water body and in which CO_2 degassed and was removed from the system (as it probably existed during formation of the evaporate). Assuming that the dissolved inorganic carbon was then converted to CH_4 in the sediments and applying fractionation factors for fermentation the modelled curve follows very well the trend of the measured CH_4 values in the evaporite sequence with values around -45 ‰ in the beginning of the evaporation process and >+20 ‰ at 95 % evaporation. These results strongly indicate that the gases are of primary origin, having formed during diagenesis of the salt layer. This again gives strong evidence that the gases are fixed in the salt formation, no migration and also no interaction with external fluids has occurred since the Permian demonstrating the long-term integrity of the salt formation for more than 250 million years [8].

In the Braunschweig-Lüneburg formation nearly no gas inclusions could be detected and also gas contents at the grain boundaries are low [9]. The authors conclude that mechanical recrystallisation caused by tectonic stress is responsible for gas release, indicating that an important prerequisite for the integrity of the formation, at least for the absence of gas migration, is that the formation was not significantly exposed to high tectonic stress.

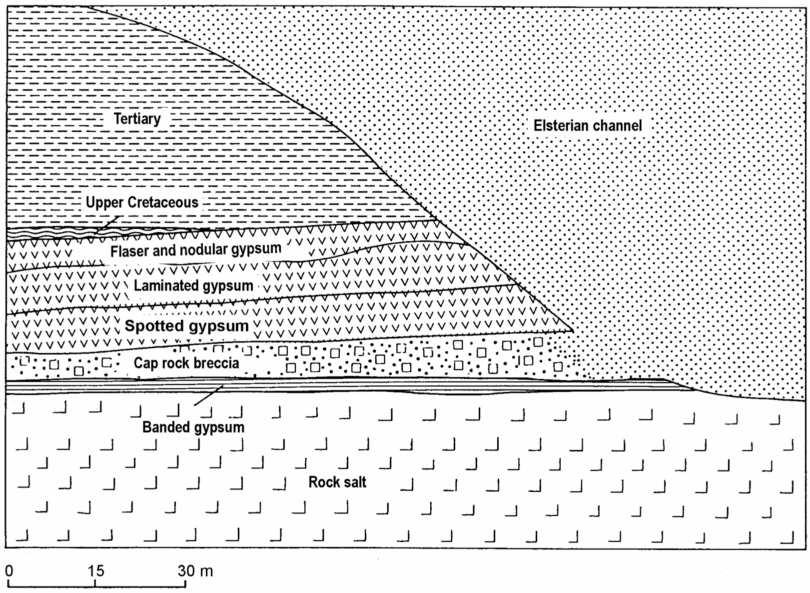
A scenario which has to be addressed because it might lead to radionuclide release in the very long term is the subrosion scenario. This scenario considers the process of salt dissolution from the top of the salt surface which might lay open the repository area in very long time scales. During the subrosion process low solubility minerals such as anhydride, gypsum or clayey material become enriched. This material builds the so-called cap rock of the salt dome. A specific situation occurs at the Gorleben site, where a dating of the recently build cap rock layer is possible which is a prerequisite for derivation of subrosion rates.

In an analogue study at the Gorleben site by the German Geological survey a detailed investigation of the cap-rock material by analysis of cores from 49 boreholes was performed [10]. In Figure 3 the typical stratigraphy of the cap rock on top of the Gorleben salt dome is shown. The cap rock consists of five different layers. The dissolution of salt occurred from top of the salt dome. The highly soluble salt is removed and less soluble gypsum layers remain. Therefore, the age of the layers decreases with depth, i.e. the oldest cap rock layer is the flaser and nodular gypsum. The layer with the so-called cap rock breccia is exceptional, since it was not formed by the subrosion process. The distribution of the cap rock breccia and its composition are clear evidence that it was formed in the period of Elsterian glaciation 500 000 to 300 000 years ago, simultaneously to the so-called "Gorleben Trough" due to the great load of the ice masses along the trough by which material not originating from the upper Permian was pressed into the trough.

Due to this fact it is clear that the layers above the cap rock breccia are pre-Elsterian, whereas the banded gypsum below the cap rock breccia was formed by post-Elsterian metamorphoses of the salt.

Therefore the thickness of the banded gypsum could be used as a measure for the subrosion process within the last 300 000-500 000 years.

Figure 3. **Diagram of the different layers on top of the Gorleben salt dome, the Elsterian channel, the rupture that was initiated from it and the resulting cap rock breccia [11]**



In 16 drillings no post-Elsterian banded gypsum was detected, i.e. no subrosion took place in these areas. The thickness of the banded gypsum varies between 0 and 40 m. From the measurement taken from 49 boreholes, subrosion rates have been derived taking into account the different content of low soluble material in the salt of each bore core. Assuming a time frame of 300 000 years for the formation of the banded gypsum layer, an averaged subrosion rate of 0.04 mm/a was estimated [11]. This is in good agreement with similarly low elevation rates of the salt dome in the range of 10-20 m in one million years, which have been derived from the quantitative analysis of the development of the salt dome in different geological eras [12]. As a consequence, only a very small part of the host rock barrier is expected to be affected by subrosion within 1 million years, since the repository will be sited at least 600 m below the salt surface.

In case of radionuclide release into the overburden, the quaternary and tertiary sediments of the overburden represent an additional barrier for radionuclide transport. In order to understand the long-term behaviour of radionuclides in such systems, an analogue site in the Czech Republic was studied, where uranium enrichment in argillaceous lignite-rich sediments occurred in a tertiary basin [13]. The detailed characterisation of the sediments showed that the major part of the uranium in the enriched zone exists in the reduced redox state U(IV) and the low $^{234}U/^{238}U$-activity ratios (<1) in this fraction gives evidence that it was immobilised over very long time scales. With new µ-spectroscopical techniques the main immobilisation mechanism of U(IV) was identified, i.e. it was shown that uranium in the U(VI) state was transported and reduced on arsenopyrite-enriched FeS surfaces by oxidation of As(0) to As(V) [14]. There is clear evidence that sulphate-reducing bacteria can contribute to maintain the strongly reducing conditions in the lignite rich zone. The study also shows that the existence of high amounts of sedimentary organic matter does not mean that organic colloids play an important role in mobilisation of uranium. In the direct vicinity of the lignite-rich layers the

DOC concentration is rather low with concentrations of few mg C/l. Sorption experiments show that the distribution coefficients for U(IV) on these sediments is not affected by humic colloids at that low DOC concentrations.

This study illustrates how efficiently uranium is immobilised if specific geochemical conditions in the tertiary sediments occur. Besides its contribution to understanding of long-term processes in natural systems, an important aspect regarding the safety case is the development of methodologies to be used for site characterisation process. This comprises sampling methods for sediment cores and groundwater under strongly reducing and low-flow conditions as well as characterisation of element and mineral distributions, redox states and $^{234}U/^{238}U$-ratios in different phases by a multi method approach.

Conclusions

Approaches and data for modelling of processes on integrated level and process level are derived from laboratory and partly from field experiments. Usually a number of experiments with well defined boundary conditions are necessary. This task can not be done by analogue studies. But it is important that natural analogue studies will not be viewed in isolation. Their key role is to be complementary to laboratory studies and modelling exercises.

The application of analogues to the safety case is to some extent dependent on the time scale they cover. Analogues for short-term processes, i.e. industrial analogues, have the advantage that the uncertainty of the initial conditions are relatively low and changes of the boundary conditions are of minor importance. Therefore, they can contribute to the understanding of short-term processes and especially to the confirmation of parameters and testing of models as it was exemplary shown for the healing process of the EDZ in rock salt. In this respect they mainly support the assessment basis of a safety case. However, due to their short time scale their use is obviously restricted.

Arguments supporting the integrity of geological formations can only be derived from geological analogues. Such analogues are of special importance for the German approach, since it is the philosophy that the geological barrier should play the most important role for the isolation of the waste. Especially for a potential HLW-repository in rock salt the long-term confinement of the waste in the host rock is the central part of the safety case. Analogue studies like profiles of content and isotope signatures of gases in fluid inclusions in not tectonically stressed salt formations showing that even gases have not migrated after formation of the layers or analyses of cap rock layers of salt domes indicating very low impact of subrosion even in time frames of 1 Million years strongly contribute to the multiple lines of evidence in a safety case underpinning the strength of geological disposal.

Acknowledgement

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GEOLOGICAL REPOSITORY PROGRAMME IN THE CZECH REPUBLIC – CURRENT STATE OF THE SAFETY CASE

S. Konopásková
Radioactive Waste Repository Authority, Czech Republic

Construction of the safety case is in the stage of methodological approach. The reason is the state of site selection process that does not allow developing a proper solution with the specific local data that would be needed to describe the disposal system performance.

Formerly, deep geological repository programme has been recently divided in following principal subtasks: site selection, design activities, near field and far field performance evaluation and safety issues. In the forthcoming period, near field and far field studies shall support the construction of a safety case methodology.

Safety Case Objectives

The principal objective of the repository safety is to meet the radiohygienical criteria: individual effective dose for a member of a critical group must not exceed 250 µSv per year. Simultaneously, the dose of workers in the operational period must not exceed 20 mSv/yr [1].

Site selection process has to be carried out using the requirements of [2], especially: in the potential site there is no intention to explore the site as a source of raw materials. The site has to be relatively well accessible by transport. By the EIA law [3], the effect on accessible components of environment has to be tolerable in the terms of the law.

A principle part of the development of the safety case methodology is to achieve compliance with current methods of safety assessment recommended by IAEA and OECD/NEA. The methodology includes:

- application of data;
- application of calculation tools;
- scenarios development using FEPs methodology;
- uncertainty evaluation;
- sensitivity analysis;
- repository safety functions, dependent on time and disposition in the repository;
- safety indicators as qualitative on quantitative expression of repository safety, dependent on time and place;
- confidence building;
- evidence of meeting regulatory requirements.

The geological repository project has been approved by State Concept in 2002, but the programme started earlier, in 1993, under support of Ministry of Industry and Trade. Some special activities are still financed by Ministry of Industry and Trade, Ministry of Education and Grant Agency. At present, there is no intention to revise the existing concept.

Supporting projects

Site selection

By the State Concept, it is obligatory to identify two candidate sites till 2015. The principal objective of the choice is the long term stability of the site and minimum radionuclide transport from the barriers protected part of the disposal system. Siting activities are subject to all legislative restrictions, esp. Regulation of SONS on siting of nuclear installations. In all existing potential six sites that have been approved after regional mapping stage in April 2003, the public opinion is still negative. As it was stated before, geological activities have been stopped till 2009 and the time delay should enable to find a consensus.

In 2007, it is planned to start a tender with the aim to precise the time schedule, project activities and volume of investment that will be necessary for starting again in 2009. The exploration licence will be issued by Ministry of Environment, but still there are awaited complications.

Research and Development Projects

In the end of 2004, RAWRA decided to advertise tenders for principal research and development activities besides site selection: near field, far filed and safety assessment. The first of them – near field research – started in the second half of 2005 and continues successfully. Far field tender was two times cancelled by legal decision and it shall be advertised again in the end of this year. As a consequence, the safety assessment tender is also delayed and it will not start earlier than in the second half of 2007. But, everything stays dependent on the approval of the state budget – because of actually complicated political situation.

Background mapping for EIA

The siting activities include Environmental Impact Assessment process that will be started in the beginning of 2007 by mapping of environmental components that would be affected by repository construction. Parallel to this, RAWRA is preparing necessary documents for licensing of research territories and representative investigation of public opinion.

Long Term and Continued Projects

- **Natural analogue** study was finished in 2005. Transport parameters obtained in the project have are used to validate the GRS transport model and they also serve as input data in international programmes, esp. 6· FP and other geological repository projects.
- **Anthropogenic analogue** solved with co-operation of Prague University specialised on glass and sludge characteristics also finished and no other activities are planned for near future.
- **Anthropogenic analogue** studies following the old tunnels EDZ performance and characterisation of granites finished in the first half of last year. A new phase of the project started this year. The research concentrates on precipitated minerals and fissure system studies. Presently it is concluded that the excavated disturbed zone changes with time and

that negative changes occur to be progressive. The study is based on long term seismic monitoring.

- Czech Geological Survey maintains its activities on the **test site Melechov**. At present, we have available six boreholes; five of them are equipped with multipacker system. There are tested new research methods as borehole televiewer and radar tomography between the boreholes (provided by company BoRaTec Weimar). Thos year's objective is to describe in detail all test polygons. Next year, there will be collected data for the far field studies as a support of the safety case by inputs from a real site.
- **P&T**. The research is supported in three areas: pyrochemistry, hydrochemistry and reactor physics. P&T is considered to be a parallel to the geological repository programme that will facilitate RAWRA to construct a geological repository in the case that the advanced technology is available. RAWRA and NRI participate in the Red-Impact project.
- RAWRA supports NRI who is participating in **Grimmsel project in the part LTD** (long term diffusion) in fracture granite media.
- Czech Technical University finished the activities in **Febex project** and dismantling of Mockup experiment. There are collected data, namely in the field of mineral chemical properties, rheology, corrosion and microbiology. The project is solved in co-operation of universities.
- NRI and two universities co-operate in the **EBS project** co-ordinated by SKB.
- RAWRA closed the project with POSIVA on the **research of montmorillonitic clays** and bentonite – the final report in English will be soon available.

Reference project

Till today, safety case is constructed for an actualised version of reference project that provides more options of design and engineered barrier system including materials used for containers construction, backfill and sealing.

There were finished more studies evaluating the inventory and properties of spent fuel.

Figure 1. **Spent fuel composition in the terms of mass ratio**

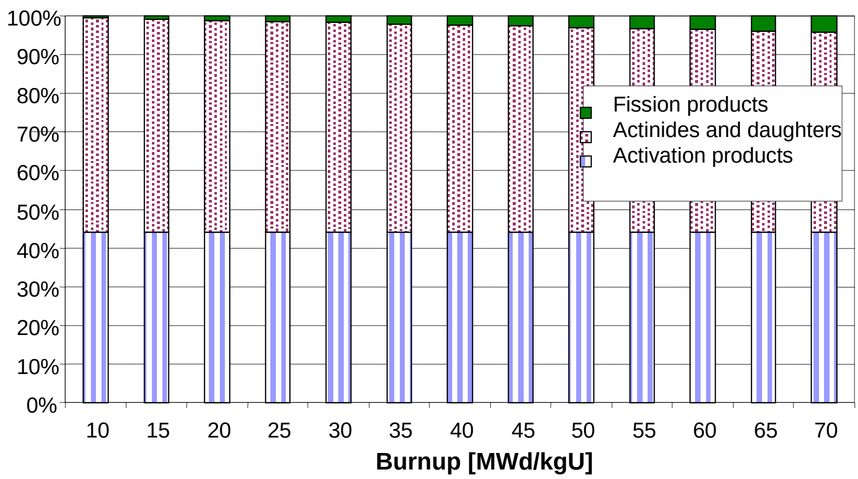


Figure 2. **Activity of the spent fuel [Bq]**

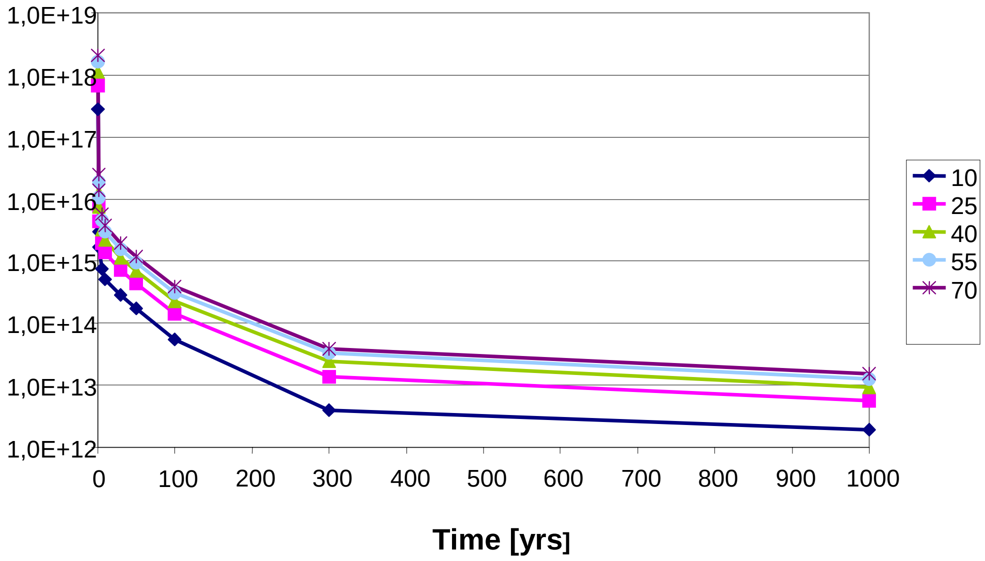
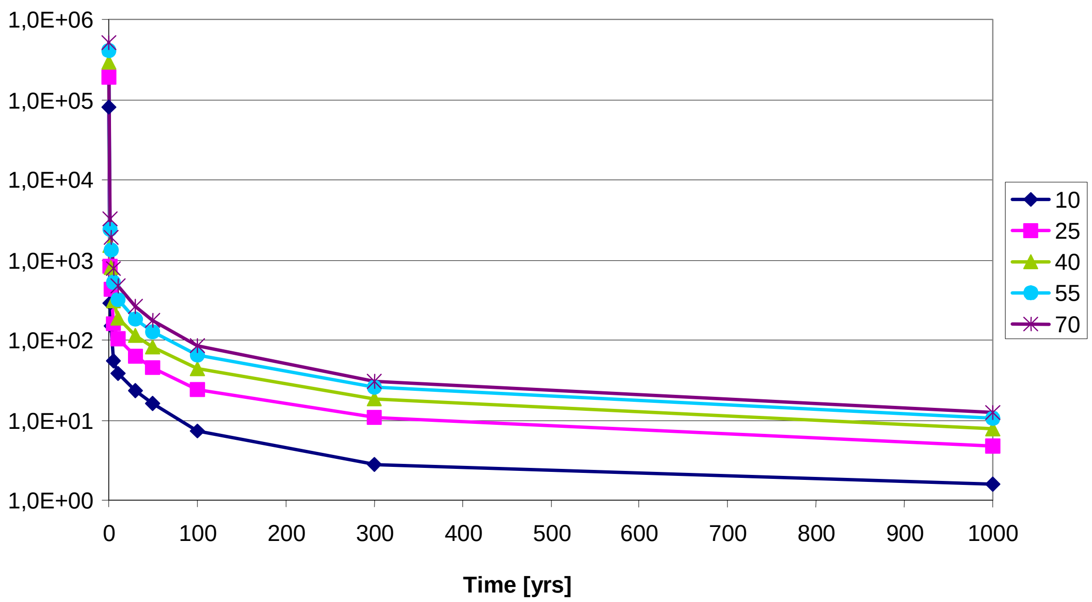


Figure 3. **Heat production [W]**



Basic sensitivity studies of migration of radionuclides in the near field and far field are available as a Test Case of the reference project [4], using unit activity 1.10^{12} Bq to calculate the relative impact of the radionuclides. Biosphere calculations in the frame of BIOPROTA project are in progress. Modelling approach and data application have been verified by comparative studies.

Figure 4. **Relative volume activity of water in the repository [Bq]**

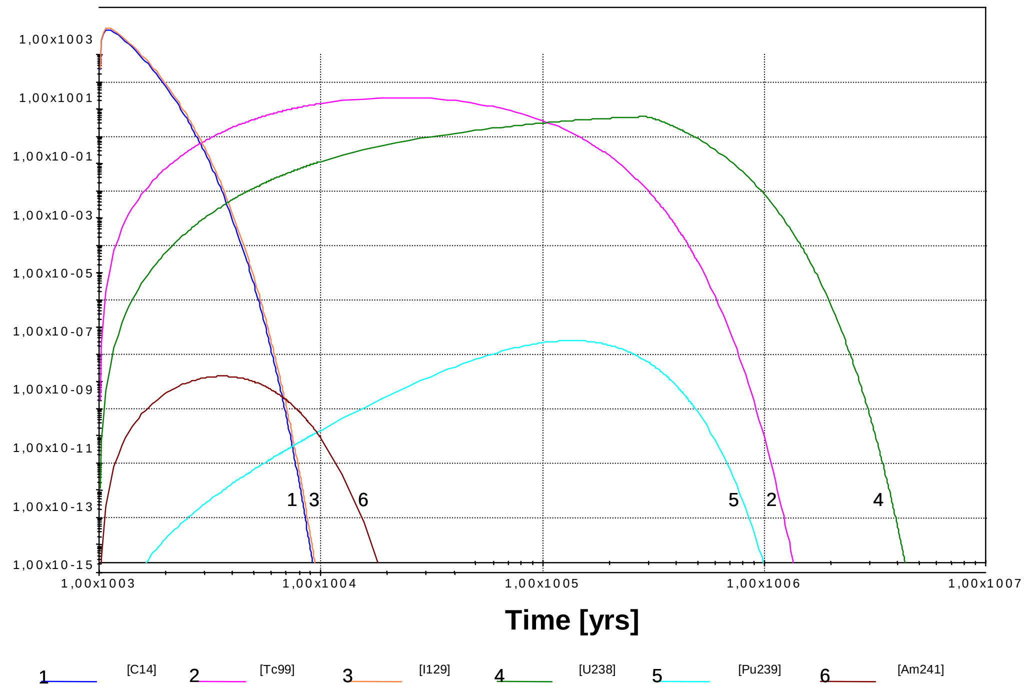


Figure 5. **Release from the repository [Bq]**

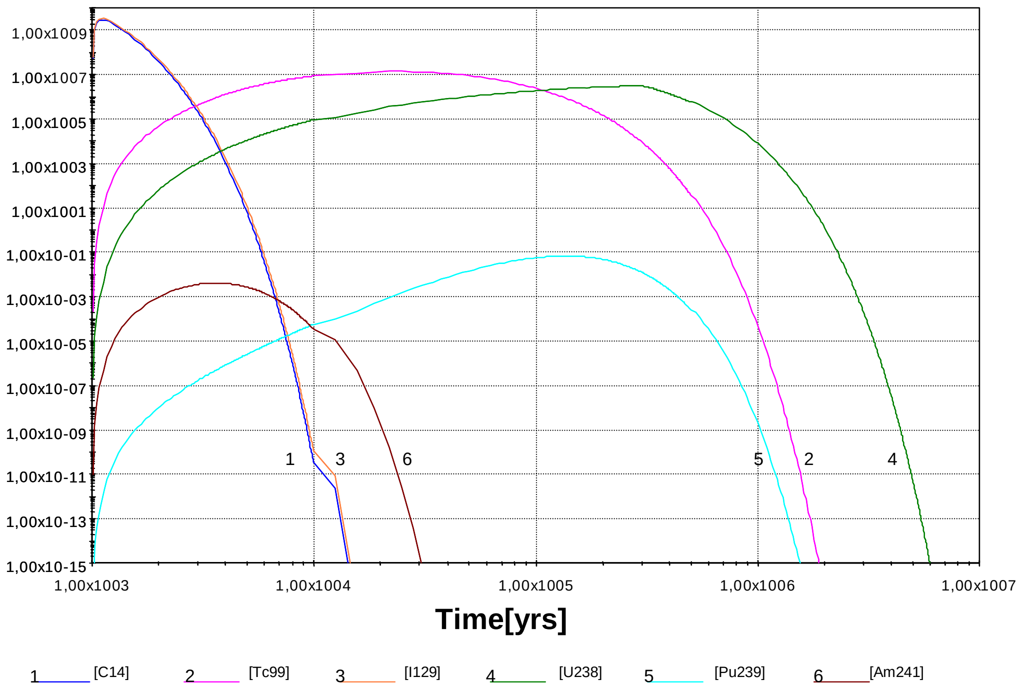
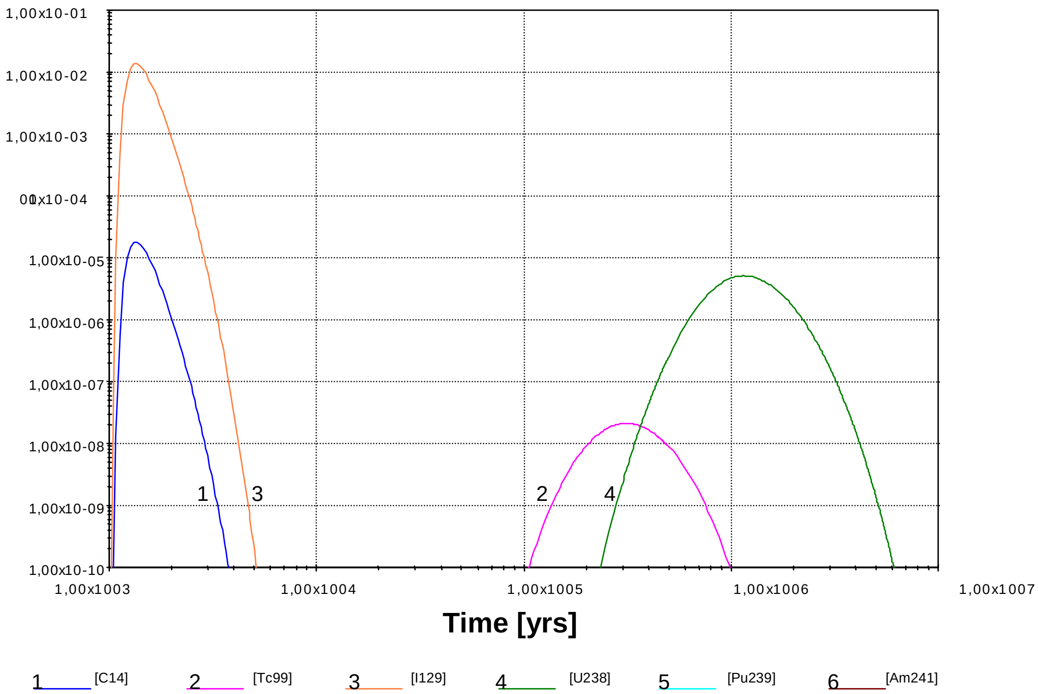


Figure 6. **Effective dose [Sv/yr]**



Near Field Studies

Near field research proceeds more than one year and the principal activities are:

- Spent fuel and high level waste degradation (emphasis to corrosion) and release rates evaluation.
- Sorption processes on bentonite, pore water studies, technetium data collection.
- THMC models, esp. for stress evaluation and following rheological stability; degradation processes in near field and subsequent modelling.
- Excavated disturbed zone thermo elastic model, qualitative studies on erosion of granites.

Near field studies concentrate on choice of materials that could be applied for the repository construction. Ongoing research provides the information on data usable in safety calculation, mostly obtained from migration experiments, corrosion rate studies and material thermal stability evaluation. These data are relevant to safety case methodology development and will be included in safety calculations assessing the near field performance with respect to near field safety functions.

Inventory calculations and thermal loading of spent fuel are continuously actualised with respect to the up-to-date knowledge concerning power plants operation and future decommissioning data.

In 2005, there was started a 3 years project that will include a basic information on barriers materials and initial study of migration parameters in the time frame corresponding to the duration of safety function in the near field. Near field studies are connected to the up to date knowledge obtained form NFPRO project.

- Biosphere – no special requirements except of those identified by EIA.
- Safety Assessment – transparency, confidence, real system description, performance of subsystems, safety indicators qualification and quantification.

Far Field Studies Objectives

Far field tender shall be called in the beginning of 2007. The project will last for about three years and its concern is to develop a method for studying and description of far field migration process with respect to the geochemical potential of a granite host structure. The support for near field decision process has to be defined considering potential effect of hydrogeology on repository performance. At this stage of the project, the far field shall be relatively simple to describe, chemically compatible to near field to provide buffer, long term stabile, it shall have just a low number of fractures, no faults. Groundwater travel times have to be as long as possible.

Confidence Building

Confidence building is presently built on natural and anthropogenic analogue studies. Participation in international programmes should have to increase the credibility of adopted decisions that are continuously carried out in the safety case construction process. Czech institutions that are involved in the safety case and environmental impact assessment activities take part in more IAEA, NEA OECD, EC and other international projects.

Expected results

The goal is to finish a “Safety Case” document for a reference disposal system, using all existing options in near field solution and with support of data of reference host structure. Simultaneously, there have to be defended use methods, reasoning and analyses including used calculation tools and input data.

Using the results of uncertainty and sensitivity analysis, the intention is to identify and possibly quantify safety functions of the repository and to define the safety indicators of repository performance.

As a conclusion, there will be elaborated a recommendation for repository design, EBS system and performance criteria for site selection process. A proposed solution has to be robust in relation to the repository safety.

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APPLICATION OF ARCHAEOLOGICAL ANALOGUES FOR A REPOSITORY SAFETY CASE: ARGUMENTS SUPPORTING THE WASTE CONTAINER LIFETIME

H.Yoshikawa, K. Ueno, M. Yui
Japan Atomic Energy Agency, Japan

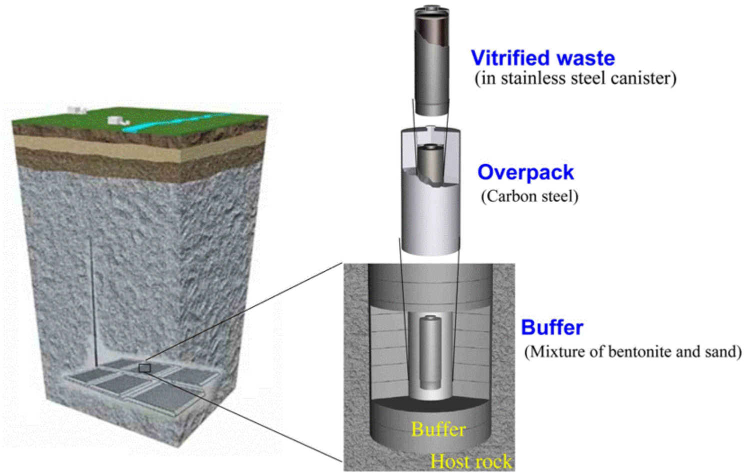
Abstract

In the Japanese HLW safety case, a carbon-steel container (overpack) was designed to have a 1 000 year lifetime, based on a corrosion allowance of 40 mm derived from laboratory data obtained under anaerobic conditions. Analogue studies have been conducted to increase confidence in the robustness of this design basis. Using X-ray computer tomography (X-CT) to measure corrosion rate, around 40 samples of archaeological iron artifacts were analysed; these were found at ancient Japanese monuments and had been buried underground for periods of several hundred to one thousand years. The corrosion rates are more than one order of magnitude less than the design allowance of 40 mm/ka, which supports the argument that the designed corrosion allowance is conservative.

Introduction

The Japan Nuclear Cycle Development Institute, the predecessor of the Japan Atomic Energy Agency, published the "Second Progress Report on Research and Development for the Geological Disposal of HLW in Japan" (H12) in 2000 [1]. This formed the technical basis for Japanese policy on geological disposal of vitrified high-level waste (HLW) arising from reprocessing of spent fuel. A multi-barrier system was adopted for the Japanese HLW disposal concept (Figure 1) [2]. The candidate overpack material is carbon-steel. The lifetime of the overpack is assumed to be at least 1 000 years, which precludes perturbations from radiogenic heat and radiolysis in the analysis of radionuclide dissolution and migration through the engineered barrier system (EBS). Some investigations of the corrosion lifetime of the overpack, based on experimental data, are contained in the report as part of the H12 safety case for a generic Japanese HLW repository [1]. The carbon-steel overpack was designed with a 1 000 year lifetime, based on a corrosion allowance of 40 mm derived from laboratory data obtained under anaerobic conditions. However, more evidence was required to confirm the long-term stability of the overpack. Analogue studies have therefore been conducted to increase confidence in the robustness of this design basis, in parallel with laboratory corrosion experiments over extended periods (several years or more). Relevant sample materials for investigating long-term iron corrosion can be found in several ancient monuments. Data on corrosion rates of archaeological iron artifacts buried underground for several hundred to one thousand years are very useful in this respect. This paper presents data on the thickness of the corrosion layer measured non-destructively using X-CT and on the main components of rust analysed by X-ray Diffract meter or destructive chemical analysis for some samples [3]. Based on this study, the level of confidence in the engineered barrier system performance is compared with that based on results from laboratory corrosion tests.

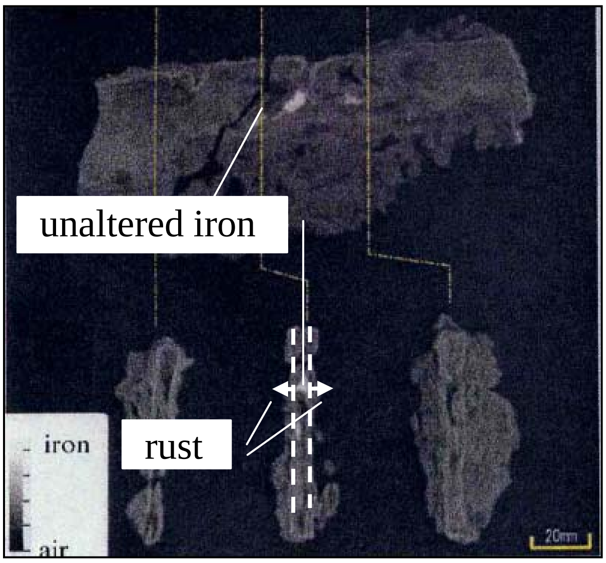
Figure 1. **Disposal concept in Japan: multi-barrier system (modified [2])**



Experimental

Iron artifacts excavated from ancient monuments are important for historical science studies. Since a non-destructive analytical method was required to obtain information on the corrosion conditions of such artifacts, we used a high-power X-ray computer tomography system (HiXCT-6M, Hitachi Ltd.), with an X-ray energy of 6 MeV and an X-ray penetration depth in iron samples of 280 mm. The two-dimensional X-ray image could be analysed to determine differences in material density. It is easy to detect remaining iron and rust layers in the samples. Quantitative analysis of the extent of corrosion was carried out based on the thickness of the rust layers. Figure 2 shows the density distribution obtained by X-ray CT imaging for a typical sample corroded under oxidising conditions. The white-colored area indicates a high density zone (8 g/cm3, typical of iron) and shows the unaltered iron material; the gray area indicates decreased density (4g/cm3, typical of rust). It is thus easy to identify rust in the sample, to measure the thickness of the rust layer and to estimate corrosion depth. The detailed experimental method was reported in a previous paper [3].

Figure 2. **Density distribution obtained by X-ray CT imaging**

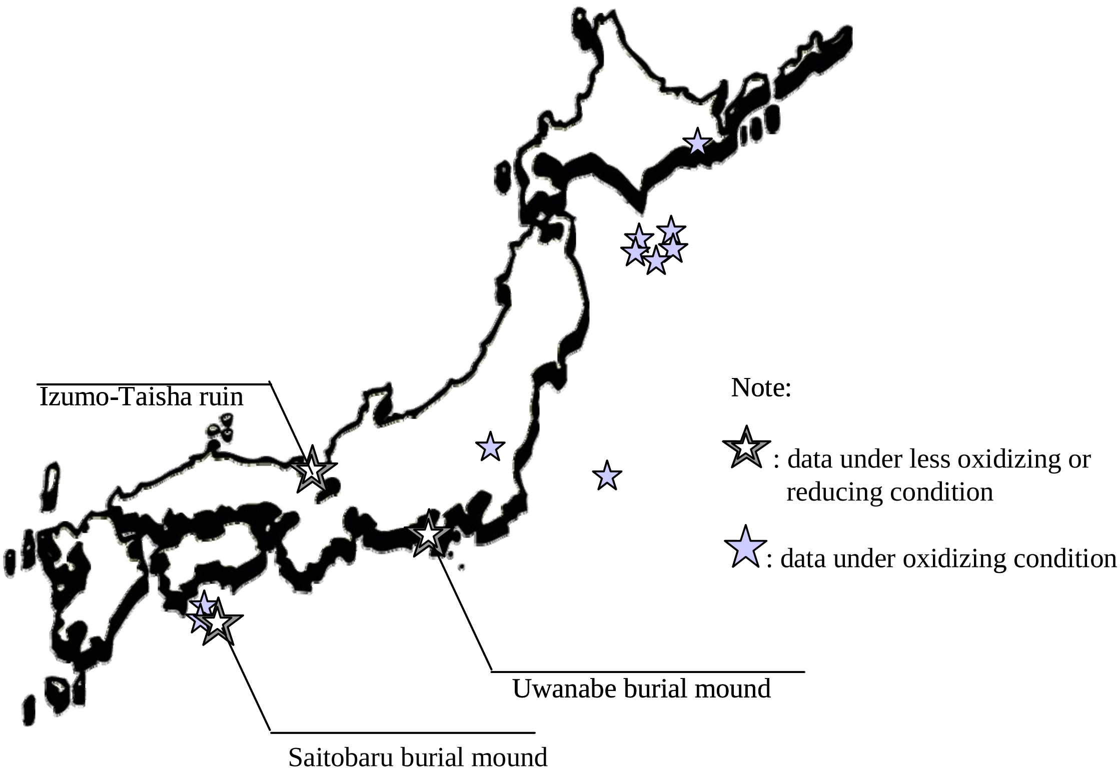


Samples

Around 40 archaeological samples excavated from 13 ancient monuments were analysed using X-ray computer tomography. The iron samples included weapons, nails and axes and their sizes varied up to about 30 cm. The excavation locations of these samples are shown in Figure 3.

Using X-ray computer tomography, both samples in which some original iron remains and samples which have corroded completely can be identified. Samples of both types were occasionally excavated from the same location, where they had been subject to the same burial period. Differences in the samples reflect differences in the burial environment, namely the presence of groundwater or redox conditions. Not all these data are described in this paper, because most artifacts are considered to be examples of corrosion under oxidising conditions in the soil near ground surface or above the water table. Amongst the samples, there are rare examples of corrosion under less oxidising or reducing conditions, which are regarded as being relevant for HLW disposal conditions. The locations where the samples were obtained were the Izumo-Taisha ruin (Shimane Prefecture), the Saitobaru burial mound area (Miyazaki Prefecture) and the Uwanabe burial mound (Nara Prefecture). These areas are indicated in Figure 3 by double stars (☆).

Figure 3. **Locations of ancient monuments providing iron samples for analysis**



Results and discussion

Some representative examples - iron artifacts which were excavated from the Izumo-Taisha ruin dating from the Kamakura age (1185-1333AD) – are described in this paper. They include belts, nails and axes made of iron that were excavated in 2000. Although only a small proportion of the original iron remained in the belts, most of the iron still remained in the axes. These axes are rare cases of artifacts being excavated from clay soil containing groundwater under slightly oxidising conditions at a depth of 2 m below ground surface (left side of Figure 4). The iron axes were excavated from bottom of the big wooden column (around 7m). The photograph on the right side of Figure 4 is vertical view of the excavation. The black circular arc at the lower part of the photograph is a part of wooden column. An arrow shows the place where the color of the rock surface changes, indicating the presence of a redox front. Studies have shown that the change in color is caused by redox reactions of manganese [4]. Dating of wood samples, which were excavated simultaneously, was also carried out by accelerator mass spectrometry analysis to provide an estimate of the duration of burial [4]. The result of the dating was 1228 ± 13 AD, indicating that these samples had been buried for about 750 years. This is consistent with the age of the Izumo shrine. The X-ray CT image of the excavated axe is shown in Figure 5.

Figure 4. **Excavated iron hand axe buried in clay soil (left) [5]: and redox front of groundwater (right) [4]**

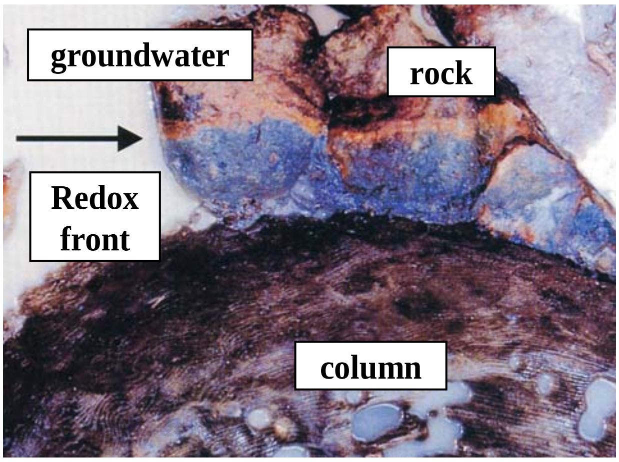
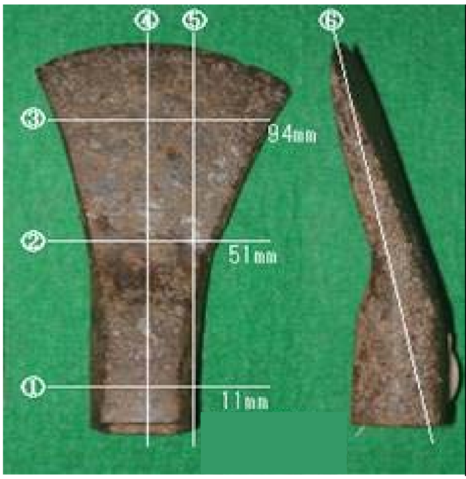


Figure 5. **X-ray CT image of excavated iron hand axe**



The image on the left in Figure 5 is a photograph of the outline and that on the right is the X-ray computer tomography image. The surface of the axe appeared dark brown and the result of analysing the density of this material indicated that it was magnetite rust.

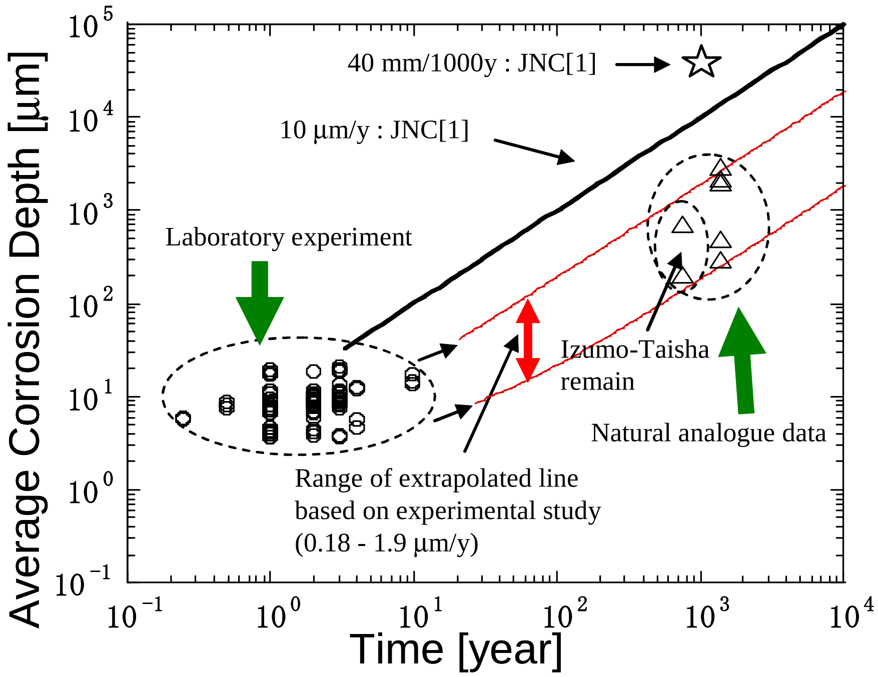
From the X-ray computer tomographic image, it is clear that the thickness of the rust layer is very small and that the axe is still mostly unaltered iron. The corrosion depth calculated from the thickness of the rust was 0.2-0.67 mm. This is compared with some other corrosion data in Table 1. The corrosion rates for the axe are an order of magnitude lower than those for samples corroded under oxidising conditions, such as belts and nails. Different amounts of rust were found on these relics, although they were excavated from similar settings and were buried for the same time period (around 750 years). It is clear that the difference in the burial environment greatly affects corrosion rate, even for relics excavated very close together.

Table 1. **Corrosion depth for excavated iron relics from the of Izumo-Taisha ruin**

Sample no.	Relic	Years buried	Color of surface	Corrosion condition	Total corrosion depth (mm)	Corrosion rate (mm/y)
1	axe	760	black	slightly corroded	0.2	2.6x10-4
2	axe	760	black	slightly corroded	0.67	8.9x10-4
3	nail	730-752	black and light brown	partially corroded	2.0	2.6x10-3
2-1	belt	730-752	light brown	completely corroded	6.3	>8.6x10-3
2-2	belt	730-752	light brown	completely corroded	6.3	>8.6x10-3
2-3	belt	730-752	light brown	iron remains	3.5	4.7x10-3
2-4	nail	730-752	light brown	iron remains	5.0	6.8x10-3
2-5	nail	730-752	light brown	completely corroded	2.8	>3.8x10-3
2-6	nail	730-752	light brown	completely corroded	2.8	>3.8x10-3
2-7	nail	730-752	light brown	completely corroded	4.0	>5.4x10-3

The data for the samples that seemed to have corroded under less oxidising or reducing conditions are shown in Figure 6, together with data from laboratory experiments. The data from laboratory experiments include previously reported values [6] plus more recent corrosion measurements for corrosion periods of 10 years. By extrapolating the laboratory data, the predicted carbon-steel corrosion after 1 000 years is represented by the star in Figure 6. This prediction is at least an order of magnitude higher than the corrosion observed for the iron artifacts, indicating that the corrosion estimate for the overpack in the H12 report is conservative.

Figure 6. **Natural analogue data on corrosion of iron artifacts compared with laboratory data and predicted corrosion in the H12 report**



Conclusions

Natural analogue studies have been conducted to increase confidence in the robustness of the H12 overpack design, in parallel with corrosion experiments over extended periods (several years or more). Using X-CT, about 40 samples of archaeological iron artifacts found at ancient Japanese monuments after being buried underground for several hundred to one thousand years were analysed. Some reliable samples which corroded under slightly oxidising or reducing conditions were obtained from 3 monuments. The thickness of the corrosion layer was measured non-destructively using X-CT. The measured corrosion rates are more than one order of magnitude less than the design allowance of 40 mm/ka, which supports the argument that the designed corrosion allowance is conservative. In particular, some samples excavated from clay layers can be considered to have been under reducing conditions for about 1 000 years and were corroded, at most, by 0.2-0.67 mm, which is two orders of magnitude less than the design allowance. Such archaeological studies can provide useful evidence for the long lifetime of the overpack, which is an important element in making a robust safety case.

Acknowledgement

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SAFETY CASES FOR RADIOACTIVE WASTE DISPOSAL FACILITIES: GUIDANCE ON CONFIDENCE BUILDING AND REGULATORY REVIEW IAEA-ASAM CO-ORDINATED RESEARCH PROJECT

M. Ben Belfadhel
Canadian Nuclear Safety Commission, Canada

D.G. Bennett
Consultant, United Kingdom

P. Metcalf
International Atomic Energy Agency, Austria

V. Nys
Association Vincotte Nucléaire, Belgium

W. Goldammer
Consultant, Germany

The IAEA has been conducting two co-ordinated research programmes (CRPs) projects to develop and apply improved safety assessment methodologies for near-surface radioactive waste disposal facilities. The more recent of these projects, ASAM (application of safety assessment methodologies), included a Regulatory Review Working Group (RRWG) which has been working to develop guidance on how to gain confidence in safety assessments and safety cases, and on how to conduct regulatory reviews of safety assessments. This paper provides an overview of the ASAM project, focusing on the safety case and regulatory review.

Background

In 2002, the International Atomic Energy Agency (IAEA) launched a Co-ordinated Research Programme (CRP) entitled "Application of Safety Assessment Methodologies for Near-Surface Radioactive Waste Disposal Facilities" (ASAM) [1]. ASAM has built on the experience gained within the previous ISAM (Improved Safety Assessment Methodology) project [2] by placing emphasis on the application of the ISAM methodology to practical problems of interest associated with real disposal facilities. The prime objectives of the ASAM project are to explore practical application of the ISAM methodology to a range of near-surface disposal facilities for a number of purposes, such as development of design concepts, safety reassessment and upgrading of existing facilities; and develop practical approaches to assist regulators, operators and other specialists reviewing safety assessments.

The ASAM work programme is being implemented through five working groups. Three application working groups are dealing with the application of the ISAM methodology to reassessing the safety of existing disposal facilities; assessing the disposal of disused sealed sources and other heterogeneous radioactive wastes, and assessing disposal of mining and minerals processing waste and other waste with an enhanced content of naturally occurring radionuclides. Two cross-cutting working groups are dealing with various common issues related to safety assessments such as regulatory reviews and confidence building, the assessment of disruptive events (e.g. human intrusion) and the performance of engineered barrier systems.

This paper provides an overview of the work being carried out by one of the cross-cutting working group, the Regulatory Review Working Group. This working group is composed of approximately 21 participants, mainly regulators, from 15 countries. The two main objectives of the RRWG are the development of systematic guidance on the regulatory review of safety assessments, and the development of practical guidance on how to document and gain confidence in safety cases for near-surface radioactive waste disposal facilities [3,4]. Although the focus of the working group is on near-surface disposal facilities, the guidance being developed is generic and in general applicable to deep geological disposal facilities.

Role, Components and Structure of the Safety Case

Safety assessments and the supporting scientific evidence form the backbone of demonstrating long-term safety of radioactive waste disposal facilities. However, because of the very long timeframes involved, decisions on long-term safety have to be made in the presence of uncertainties such as those associated with the characterisation of natural systems and the evolution of the waste facility. Because of these uncertainties, it is not possible to demonstrate the safety of radioactive waste disposal facilities in an absolute way, and something more than just computational compliance with numerical safety targets is required. Human judgment on the choice of site and facility design, the overall approach to and conduct of the safety assessment and the depth and quality of all safety related work is an integral component of the decision-making process and an appropriate objective for the regulators and other stakeholders is to expect the implementer to provide "reasonable assurance of safety".

Providing "reasonable assurance" that a disposal facility will, in the short- and long-term, perform in a manner that protects human health and the environment should be achieved through the development of a safety case, which puts the findings of the safety assessment into a broader context with other factors and considerations that are relevant to the decision-making process and important to the stakeholders involved. In other words, the safety case should explain why the intended audience should have confidence in the acceptability of the disposal facility.

The IAEA and NEA define the safety case as an integration of arguments and evidence that describe, quantify and substantiate the safety, and the level of confidence in the safety, of the radioactive waste disposal facility [5,6]. Based on international experience and current practices in many countries, the confidence building arguments that can be made when assembling a safety case should include both technical, economic, societal and managerial arguments. With regards to the technical elements, the following should be included in the safety case:

- A well-documented, traceable safety assessment, including discussions on how confidence was established within each stage of the assessment and in the overall safety assessment methodology, approach and findings.
- A discussion on the level of confidence that exists in the robustness, understanding and reliability of the disposal system and in the underlying engineering and science.
- A description of the uncertainties that exist, an explanation of how these uncertainties have been quantified and/or otherwise assessed, and a plan or programme for reducing the key uncertainties and resolving any remaining issues.
- A discussion of alternative waste management options to justify the selected option (storage or disposal site, safety strategy, risk management approach etc.).

In addition to these elements, other aspects are important for building trust and confidence in the credibility of the implementer as well as in the regulatory review and decision making process. These elements may include:

- A demonstration that the audience for the safety case should have confidence in the management frameworks and competence of both the implementer and the regulator.
- A demonstration that the safety case has been developed via a transparent process with appropriate independent review and stakeholder involvement.

Although some countries do not use the term "safety case" in a formal way for near-surface disposal facilities, the approaches and processes used to demonstrate safety for such facilities are compatible and, in essence, similar to the safety case concept used for deep geological disposal, and most of the arguments supporting the safety case are embedded within more or less explicit regulatory requirements. In many ways the safety case can be viewed as a tool for integrating a wide range of information, demonstrating, promoting and communicating confidence in the long-term safety of a radioactive waste facility. The content and structure of a safety case is greatly influenced by country-specific and site-specific legislative and regulatory requirements and local concerns.

The development of a safety case is an iterative process that spreads over a long period of time and evolves with the development of the disposal programme and facility. For this reason, the arguments included in the safety case may carry different weight and importance. The level of scrutiny that they are subjected to by stakeholders may vary over time, depending on the development stage of the facility and the regulatory decision that is under consideration (e.g. early planning and stakeholders consultation stages, environmental assessment, licensing etc.).

Structuring and Documenting the Safety Case

Structuring and documenting the safety case presents a number of challenges because the target audience is composed of a wide range of stakeholders with different needs, expectations and concerns. These include regulators, political decision makers, the general public, special interest groups and others. Another challenge is related to situations where there are complex legal and regulatory requirements, possibly involving multiple regulatory agencies with different regulatory processes, and where multiple levels of documentation are required throughout the development stages of a radioactive waste facility (Environmental Impact Assessments, Preliminary Safety Reports, Safety Analysis Reports for licensing etc.). Given these challenges, there is no universal structure for documenting the safety case. Whatever documentation structure is adopted, there are key attributes and considerations that should be considered throughout the documentation process. These include the following [2,5,6,7]:

- All documents produced in the context of the safety case, whether for regulatory approval, for information or promotion should convey a consistent message about safety. In other words, the story should remain the same and there should not be changes to suit the perceived expectations of particular audiences.
- The foundation of the safety case should always be sound scientific evidence and arguments, established using technical experience and analyses.
- The documentation should be transparent by making the information readily available to stakeholders, being clear and understandable, and by clearly presenting the justifications and rationale behind key assumptions.
- The documentation should provide a traceable record of how decisions have been made, and should include a clear, accurate and precise referencing system.

- The documentation should engender trust in the safety assors by openly acknowledging uncertainties and limitations as well as their safety implications.

While recognising that there are many possible ways of structuring and documenting a safety case, there are common elements that should be considered and clearly documented. These include: high-level summaries, clear descriptions of the context for the safety case, of the safety strategy and approach, of the safety assessment, and of stakeholder involvement and consultation processes.

Regulatory Review of Safety Cases and Safety Assessments for Radioactive Waste Disposal Facilities

Regulatory reviews of safety cases and safety assessments are primarily conducted to assist regulatory decision making on the licensing or authorisation of a radioactive waste disposal facility. The conduct of a high-quality review enhances confidence in the credibility of the regulator, in the review findings and in any associated regulatory decisions. It is therefore important that the process by which regulatory authorities conduct such reviews is systematic, logical and defensible (open and transparent), based on clear safety criteria, and leads to clear and logical decisions.

In this context, the ASAM Regulatory Review Working Group is developing practical review guidance that covers a number of aspects [3]. These include the attributes and objectives of the review; the overall management and conduct of the review; the management of review comments, conflict resolution, and reporting the findings of the review. Guidance on specific issues and challenges such as dealing with regulatory uncertainty; conducting a review with limited resources, and stakeholder involvement during the review process is also being developed. A summary of the guidance being developed for selected topics is provided in the following sections.

Objectives of the Regulatory Review

The objectives of regulatory reviews of safety assessments and safety cases will need to take account of the status of the disposal facility and the associated assessment context, but will typically include:

- Ensuring that the disposal facility will not cause unacceptable adverse impacts on human health safety, and on the environment now and in the future.
- Determining whether the safety assessment has been conducted in an acceptable manner (quality, level and depth) and whether it is fit-for-purpose.
- Verifying that the results of the safety assessment and the assumptions on which the assessment and the wider safety case are based, comply, or are in accordance, with accepted radioactive waste management principles, and regulatory requirements and expectations.
- Ensuring that relevant measures and contingencies to mitigate possible potential effects have been identified and considered, and that adequate follow-up plans for their implementation have been developed.
- Ensuring that uncertainties have been adequately identified and, to the extent possible, constrained and quantified.
- Helping regulators understand the issues that are most important to safety.

Attributes of the Regulatory Review

There are a number of key attributes that may influence the quality and success of a regulatory review. These include:

- The regulatory review should be independent from reviews undertaken by the implementor as part of the process to develop the assessment.
- Regulatory requirements and expectations, including the criteria against which safety will be judged, should be clearly defined and explained to the proponents/implementers and other stakeholders.
- The scope of the review should be clearly defined and the review team should not be unduly influenced by considerations that are outside the scope of the review. Any such considerations may be taken into account in a broader context by decision makers, together with the safety case review findings.
- The regulatory review process should be structured and traceable with clearly defined roles and responsibilities and decision-making steps.
- The regulatory review should be conducted using a level of resource that is adequate and commensurate with the level of complexity of the safety assessment and the risks associated with the facility under consideration.
- The overall process of regulatory review should include a stakeholder consultation framework with well-defined consultation steps, rules of procedure and decision making.
- The regulatory review should document the rationale for judgements as to whether or not the arguments presented in the safety assessments or safety case are adequately supported by the underlying science and technology, and whether those arguments are in accordance with regulatory requirements and expectations.

Fulfilling the expectations raised by these attributes can represent a significant challenge in itself. For example, regulatory authorities should possess a sufficient level of internal expertise and hands-on experience in assessing the safety of radioactive waste facilities. Furthermore, reviewers will need to consider both "hard" issues (e.g. objective, technical or scientific issues) and "soft" issues (e.g. subjective, value-related issues). Differences of opinion are typically more common and more difficult to resolve on the soft issues than on the hard issues, owing to differences in reviewer's values, experiences, judgements and expectations.

Managing the Review Process

The management of a safety case or safety assessment review is treated as a project, to which the standard principles of good project management apply. Depending on the scale of the review to be conducted, it is often necessary to establish a team of reviewers. Regulatory reviews may be conducted by the regulatory authority with or without support from external organisations, but the results of the review must be fully "owned" by the regulatory authority. The guidance being developed under this topic considers the main following aspects:

- Defining the objectives and scope of the review.
- Developing a review plan that identifies the review tasks and addresses other relevant topics listed here.
- Assembling a review team of competent personnel possessing the necessary expertise and experience.
- Defining the project schedule and allocating resources for the conduct of project tasks, including consideration of review conduct when resources are limited.

- Identifying the responsibilities of review team members and ensuring that they receive adequate training and guidance in the review method.
- Co-ordinating the conduct of the review tasks, and ensuring sufficient communication between review team members.
- Identifying early on during the review any areas of regulatory uncertainty. An example of this type of uncertainty might relate to question of whether or how to apply new environmental protection standards when upgrading the safety of old disposal facilities.
- Co-ordinating dialogue with the operator of the disposal facility, and with other stakeholders during the review process.
- Reviewing and integrating documents generated during the review process.
- Synthesis and communication of review findings.

The review procedures applied should allow the regulatory authority to demonstrate that the review of the safety assessment has been performed by competent people, and recorded in a traceable and auditable manner. Project-specific procedures might include structured approaches for documenting review comments, for specifying required competence, for specifying responsibilities and tasks in the review, for recording the status of issue resolution, and for conflict resolution. Further procedures may be necessary if the review includes tasks such as audits or independent regulatory assessment calculations.

For each regulatory review, a review plan will be required to provide guidance on procedural and technical aspects of the review. Procedural guidance might include the means of documenting the review findings. Technical guidance might include the criteria against which to judge specific aspects of the safety assessment. The RRWG document may, therefore, serve as a template from which a project-specific plan can be developed. Examples of project-specific review plans include those developed for the Drigg low-level radioactive site in the UK [8] and for the Yucca Mountain project in the USA [9].

Conducting the Technical Review

This RRWG guidance document also identifies the main components of the safety assessment to be considered during the review. The guidance follows each of the steps within the ISAM assessment methodology [2], that is, the assessment context; the system description; the development and justification of scenarios; the formulation and implementation of models; and the analysis of results. For each step, the review procedure highlights the types of statements of confidence that the safety assessment should support, and lists appropriate questions that the reviewer might ask when conducting the review. Such questions are designed to help reviewers identify potential deficiencies in the safety assessment, but they should also help safety assessors to better document their safety assessment and thereby meet regulatory expectations.

In order to assist with evaluating the safety assessment against the primary review objectives, it is common for a number of additional objectives to be specified. These may include evaluating whether the safety assessment:

- Is based on an appropriate assessment context.
- Is sufficiently complete, given the status of the disposal programme and disposal facility.
- Is sufficiently transparent in its presentation of data and information.

- Is based on appropriate assumptions and contains adequate arguments supporting the adoption of those assumptions, including assessment scenarios, models and parameter values.
- Demonstrates an adequate understanding of the disposal system.
- Clearly identifies the uncertainties associated with the understanding of the disposal system and the performance of the disposal facility.
- Includes an adequate consideration of optimisation and/or intervention.
- Has been conducted under a suitable quality assurance system.
- Defines an appropriate forward programme for improving the safety assessment, understanding of the disposal system, and control of the site.

Conducting Regulatory Reviews with Limited Resources

The conduct of effective regulatory reviews requires an adequate level of resources as it is usually a necessarily detailed and complex exercise covering a wide range of disciplines and expertise. However, the resources available to regulatory authorities in different countries vary widely according to the scale of national nuclear programmes. In many small countries the number of regulatory staff that can be dedicated to the review of a safety assessment for a particular waste disposal facility remains low and, in some cases, is limited to just one or a few individuals. While some larger countries have access to more resources, they are often faced with the same difficulties because of the larger size of their nuclear regulatory programmes and the increasing level of scrutiny by the public and other stakeholders.

Failing to provide adequate resources in the context of regulatory reviews may engender serious consequences. These include eroding public confidence in the regulator's decision and in its ability to ensure safety, decreasing the level of regulatory scrutiny and delaying consideration of applications and proposals. Another consequence is the potential for an increasing level of workload-related difficulties within regulatory organisations, which may lead to lower levels of productivity and vigilance. It is not uncommon for organisations to place very significant reliance on key individuals dealing with a particular facility over several years or more.

Addressing the issue of limited resources is challenging, and the proposed solutions are not always unanimously accepted by the various stakeholders involved, mainly because of diverging perceptions of risks and conflicting priorities. Possible approaches to address the issue of limited resources include using risk-informed reviews, use of internationally agreed generic disposal concepts and assessment tools.

Use of Risk-Informed Regulatory Reviews

An increasing number of countries (e.g. Canada, Japan, Sweden, UK, US) have developed, or are considering or developing, risk-based or risk-informed regulatory frameworks [10]. Such approaches are designed to improve regulatory efficiency by focusing regulatory scrutiny on areas where there is the greatest potential to achieve safety or environmental protection benefits.

While it is relatively easy to apply risk-informed approaches when undertaking and prioritising activities relating to regulatory compliance of operating nuclear facilities, it is more challenging to develop and apply such methods to regulatory reviews of safety cases and safety assessments for radioactive waste disposal facilities. This difficulty arises because of the prospective nature of such safety assessments and also because of the need to consider both the details of the assumptions on which the assessment is based as well as the assessment results. In regulatory review of safety assessments, the “Devil is in the detail”. It is important, therefore, that regulatory reviews are conducted to a level of detail appropriate to the technical and scientific basis of the issues being considered. A key question is, thus, how to balance the scope and complexity of a regulatory review against cost and time, whilst at the same time ensuring safety.

Another difficulty related to applying risk-informed approaches to regulatory reviews of safety cases for disposal facilities is that such approaches may prevent the regulator from developing a thorough understanding of all of the different aspects related to the assessment of the facility under review. This may be seen by the public and other stakeholders as a lack of competence even when this lack of understanding is associated with low risk issues. This could be particularly the case during public hearings and meeting when the regulator is challenged and unable to demonstrate a thorough understanding. One approach to combating this difficulty may be to consider assessment results for a range of sub-system performance criteria other than dose or risk, such as radionuclide release from the near-field, or flux through a particular engineered barrier. Regardless of the details of the approach chosen, the regulatory review plan should ensure that the general requirement on the proponent/ implementer to demonstrate a sufficiently broad understanding of the disposal system is adequately assessed and reflected by the regulatory review team across the full breadth of the safety case.

Conclusions

The work of the ASAM Regulatory Review Working Group is providing practical tools and techniques aimed at improving the safety of radioactive waste disposal facilities. The guidance on regulatory review will provide a consistent framework and approach for a review of safety assessments and, in many respects, may also be applied to the review of safety cases. The guidance on safety assessment review can also be applied at a level of detail compatible with available resources. This is particularly important for countries with severely limited resources. The guidance on confidence building in the safety case should help in addressing the particular challenges associated with the development, upgrading and remediation of near-surface disposal facilities.

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C-14 AND THE SAFETY CASE

G. Bracke, W. Müller
ISTec – Institute for Safety Technology, Germany

Introduction

A safety case as synthesis of evidence, analyses and arguments is supposed to quantify and substantiate a claim that a repository is safe after closure and concludes that there is adequate confidence [4]. A safety case is also described as a collection of arguments, at a given stage of repository development, in support of the long-term safety of the repository [3]. If there is a weak point in this supporting line because of uncertainties, additional efforts are necessary to keep the confidence of the audience in the safety case.

Carbon-14 (C-14) is one of the major radionuclides released from the nuclear fuel cycle into the environment. It has long been noted that the relatively long half-life of C-14 (5 730 years), together with its mobility in the environment and incorporation into man via the food-chain, leads to the long-term potential exposure.

Depending on the scenario, C-14 can contribute significantly to potential radiation exposure in the biosphere, so it has to be included in the safety case of a disposal facility. There are ongoing studies [1] on the consequences of releases of C-14 to particular types of environment which demonstrate difficulties and uncertainties in modelling the release of C-14. This is mainly attributed to the versatile chemistry of carbon.

The pathways of C-14 and relevant processes are discussed and assessed in the following paper. Since a conservative approach may lead to overestimation of the radiation exposure by C-14 and restrict the credibility of a safety case [4] a more realistic approach will be recommended [2] to provide additional features, events and processes (FEP) for consideration in safety cases of repositories.

Inventory of C-14

In the following the FEPs related to the inventory of C-14 are listed:

- The inventory of C-14 and other constituents of the waste is set when emplacement is finished. A build-up of new C-14 activities by decay chains or nuclear reactions is impossible in low level radioactive waste. The inventory is estimated in a way to cover all uncertainties.
- From stocktaking some waste fractions are known and the waste can be classified into organic/anorganic/metallic and non-metallic fractions. The organic fraction accounts typically for 10-30% of the low level waste. Since there is no detailed documentation of the speciation of C-14 activity in low level radioactive wastes, the C-14 activity cannot be assigned in greater detail.

- The content of C-14 and its concentration may vary within several orders of magnitude in different emplacement locations or waste packages. Table 1 depicts a range of contents and concentration of C-14 in a repository in salt rock for low and intermediate level waste. The inhomogeneity of the C-14 concentration varies typically by a factor of 10 only. Additional conservative accounting by assuming the maximum concentration is not considered necessary.

Table 1. **Content and concentration of C-14 in emplacement chambers**

Emplacement chamber	C-14 activity (10^{11} Bq)	C-14 concentration (10^{8} Bq/Mg)
lowest	0,02	0,05
range	0,5 – 5	1-10
highest	10	35

Release of C-14 from waste fractions

It is generally assumed that the C-14 inventory is bound to organic matter. This assumption omits the fact that C-14 can be bound to solid phases such as concrete containing carbonates or metals containing carbides.

If the C-14 activity were homogeneously distributed in the waste, a significant amount (>90 %) would not contribute to the generation of gaseous methane or carbon dioxide by degradation of organic matter, which is of major concern within the safety assessment. The release of C-14 activity would be slowed down significantly in the long term.

Carbides occurring in ashes and having contact to brine will generate methane instantly. This methane would be removed quickly from the mine by ventilation before closure and would not be accounted for long-term safety.

The C-14 fraction in metals would be released within the progress of corrosion mainly as methane under anaerobic conditions. The release of C-14 to the brine and gas phase is delayed since the progress of corrosion is slow in comparison to microbial degradation of organic matter. The release of carbon from corrosion is assumed to be complete in the very long term, but the half-life of C-14 would reduce the total released activity in the long range.

The degradation of organic matter is limited and will not be complete under saline conditions as shown by experiments and natural analogues [6]. A large fraction of C-14 will remain within the deposited waste. The C-14 activity will decline according to its half-life.

The speciation of the major C-14 compounds may be different from those of the remaining organic fraction in the low level radioactive waste.

Brine Pathway

The transition of C-14 from deposited radioactive waste to the brine depends on its chemical speciation. Possible processes are:

- dissolution of superficial contamination;

- lixiviation from metals. This process is well known and has been studied on hulls from zircaloy and activated core parts from LWRs [9];
- microbial degradation, dissolution or desorption of organic compounds;
- isotope exchange of C-14 with C.

Low level radioactive waste is heterogeneous. Due to the lack of detailed information it has to be treated globally.

The rate of transition and accumulation of organic C-14 activity may correlate with the gas generation process, its rate and yield.

The effective concentration of C-14 activity in the brine depends on the geochemistry and the composition of the solution. The effective C-14 activity is lowered by precipitation of carbonates, sorption of CO_3^{2-} on backfill [8], and isotope dilution when the C-14 bearing brine is transported elsewhere [5].

The degradation of organic matter produces intermediates, which may be soluble in brine. These intermediates will be comparable to substances from standard waste dumps under anaerobic conditions and consist mainly of glucose, amino acids, fatty acids, hydroxycarbonic acids, isosaccharinic acids etc. [7] Final products will be methane and carbon dioxide. The solubility of methane in brine is low. Carbon dioxide will be one of the main components and be in equilibrium with dissolved and precipitated carbonates.

Reliable equilibrium constants are known for alkali and earth-alkali carbonates, carbonato complexes and mixed solid phases, and a nearly complete data set of Pitzer constants is available. This data is used for geochemical modelling of brines. The main compounds of carbonates occurring in a salt mine are $MgCO_3$(aq), $CaCO_3$(aq) and CO_3^{2-}. Solid phases such as magnesite, calcite and dolomite limit the concentration of dissolved anorganic carbon.

A range of concentrations which are expected in emplacement chambers assuming a closed system are shown in Table 2. The dissolved fraction of anorganic carbon represents only a small amount of the total carbon inventory. The effective concentration of dissolved carbonates may vary according to the geochemical setting and long-term stability. The major amount of carbonates is fixed in the solid phase.

Table 2. **Range of concentration and inventories of carbon and carbonates in emplacement chambers**

Speciation	**Concentration (10^{-5} mol/kg H_2O)**
C (TIC)	0,8 – 40
CO_3^{2-}	0,04 – 1
HCO_3^-	0,000001 – 3
C (Inventory)	200 000 – 3 000 000

Equilibrium processes, precipitation, sorption and gas generation apply to C-14 containing compounds as well as to other carbon compounds. The isotope exchange of C-14 containing carbonates takes place rapidly in solution by hydrogen exchange.

$$^{14}CO_3^{2-} + HCO_3^- \leftrightarrow CO_3^{2-} + H^{14}CO_3^-$$

The isotopic equilibrium of gas phase and fluid phase is achieved rapidly as well.

$$^{14}CO_2\,(g) + HCO_3^- \leftrightarrow CO_2\,(g) + H^{14}CO_3^-$$

The isotope equilibration of solid phase and solution may be delayed since the dissolution and precipitation is controlled by kinetics and accessible surfaces. The isotopic equilibration of other compounds such as hydrocarbons, fatty acids or alcohols is controlled kinetically and may be delayed.

Therefore, as a first approximation, C-14 can be assumed as homogeneously distributed in all carbon species in solution. The concentration of C-14 will be significantly lower than the total concentration of carbon. The typical isotope ratio of C-14 to total carbon in the low-level waste in our example equals 10^{-8}-10^{-9}.

When brine is transported, processes achieving an isotopic equilibrium are going on, which may delay the transport of C-14 by precipitation and dilution.

Summary

The following can be summarised:

1. The transfer of C-14 into brine is not instantaneous.
2. The transfer rate of C-14 into brine is correlated to a degradation rate of organic matter.
3. The concentration of C-14 in brine is limited due to equilibration with carbonates in the gas phase and the solid phase and the occurrence of intermediates.
4. Due to the composition of the brine and the relative amount of mineral phases, the predominant amount of C-14 will be fixed in the solid phase as carbonates.
5. Isotope exchange processes will equilibrate any C-14 activity in solution.
6. Sorption, precipitation and isotope exchange will take place when solution is transported and lower the C-14 activity in solution.

Gas Pathway

C-14 can be present in all carbon-containing gaseous compounds. In sequence of decreasing amounts, the expected gas phase consists of:

1. methane, $^{14}CH_4$
2. carbon dioxide, $^{14}CO_2$
3. cydrocarbons
4. other volatile compounds.

Therefore, the release mechanisms of C-14 from the waste to the gas phase are of interest, too.

Annual records of mine operation of final radioactive waste repositories show a release of C-14 activity by mine ventilation. Radiation exposure is of limited concern, but a continuous release of C-14 from deposited waste accounts at a minimum for a fraction of 2% within 30 years of operation. According to previous studies, 75 to 90% of the released C-14 activity is speciated as CO_2. The remaining fractions are mostly hydrocarbons. The domination of $^{14}CO_2$ can be explained by microbial degradation under aerobic conditions, which prevails in the final radioactive waste repository during the operational phase.

After closure of the repository, the release of C-14 activity will continue by microbial degradation of organic matter, corrosion of C-14-containing metals, or hydrolysis of carbides.

1. Hydrolysis of carbides, such Al4C3, will instantly release 14CH4.

$$Al_4C_3 + 12\ H_2O \quad \leftrightarrow \quad 4\ Al(OH)_3 + 3\ CH_4$$

2. In the long term, microbial degradation will be the major release mechanism of C-14 to the gas phase. According to previous studies, gas generation by microbial degradation will be lower than the total theoretical yield.
3. An additional release of C-14 activity could be possible if methanogenesis is considered. Hydrogen will be present in any final repository due to the anaerobic corrosion of metals.

$$CO_2 + 4\ H_2 \ = \ CH_4 + 2\ H_2O$$

   Methanogenesis is supported by microbial activities and by catalysis. If this reaction takes place at a significant rate, the total amount of gas will be reduced by a factor of 4. This would change the modelling of fluid transport. Due to the low enthalpy, methanogenesis is least favoured in comparison to other reactions.

Gaseous compounds generated from organic matter need not to be released to the gas phase instantly. Retention of gaseous compounds is possible. Adsorbing materials such activated carbon or waste conditioning materials such as cement and concrete may fix $^{14}CO_2$ and other compounds permanently.

Surface adsorption on backfill provides an additional mechanism for the retention of C-14 containing compounds.

The capacity for precipitation of carbonates may be increased by backfill and may fix $^{14}CO_2$ permanently.

Equilibrium processes of brine, gas phase and solid phase will limit the concentration of C-14 compounds, especially CO_2, in the gas phase, depending on the geochemical conditions set by the closure concept.

The isotopic equilibrium processes will dynamically dilute the C-14 content of the gas phase on transport when uncontaminated compartments are passed.

Oxidation of C-14-containing compounds on transport by sulphate-containing rock and backfill to CO_2 and subsequent precipitation can limit further the C-14 release from the mine.

Storage of gas-containing C-14 compounds in the mine will lower the released fraction of C-14 to the biosphere.

In summary, the following can be said:

The release of C-14 with the gas phase to the biosphere will be limited by the following processes:

1. The mobilisation of C-14 depends on the rate of degradation of organic substances to gaseous compounds.
2. Equilibrium processes limit the concentration of $^{14}CO_2$ in the gas phase.
3. Carbon and its isotopes are in exchange with solution, solid and gas phase. Isotope exchange processes will dilute the C-14 concentration on transport.
4. Sorption and precipitation will lower the C-14 content of the gas phase.
5. A fraction of the inventory of C-14 is released to the environment by mine ventilation before closure.

Considering these effects will lower the potential radiation exposure via the gas phase significantly.

Radiation Exposure

Radiation exposure by C-14 depends on site-specific scenarios. From qualitative considerations, the following can be demonstrated for conservative assumptions:

1. The long term assessment of a potential radiation exposure by $^{14}CO_2$ demonstrates in general on either pathway (brine, gas) that dose limits are easily met.
2. If C-14 is released as CH_4 via gas directly to the overlying rock and migrates after oxidation to the biosphere, a potential radiation exposure by ingestion may become relevant.

Therefore, we recommend a realistic approach to long-term safety assessments considering the individual steps within the safety case to demonstrate safety and safety margins to competent authorities and their experts and avoid dose constraints for purely fictional potential radiation exposures.

Distribution of C-14

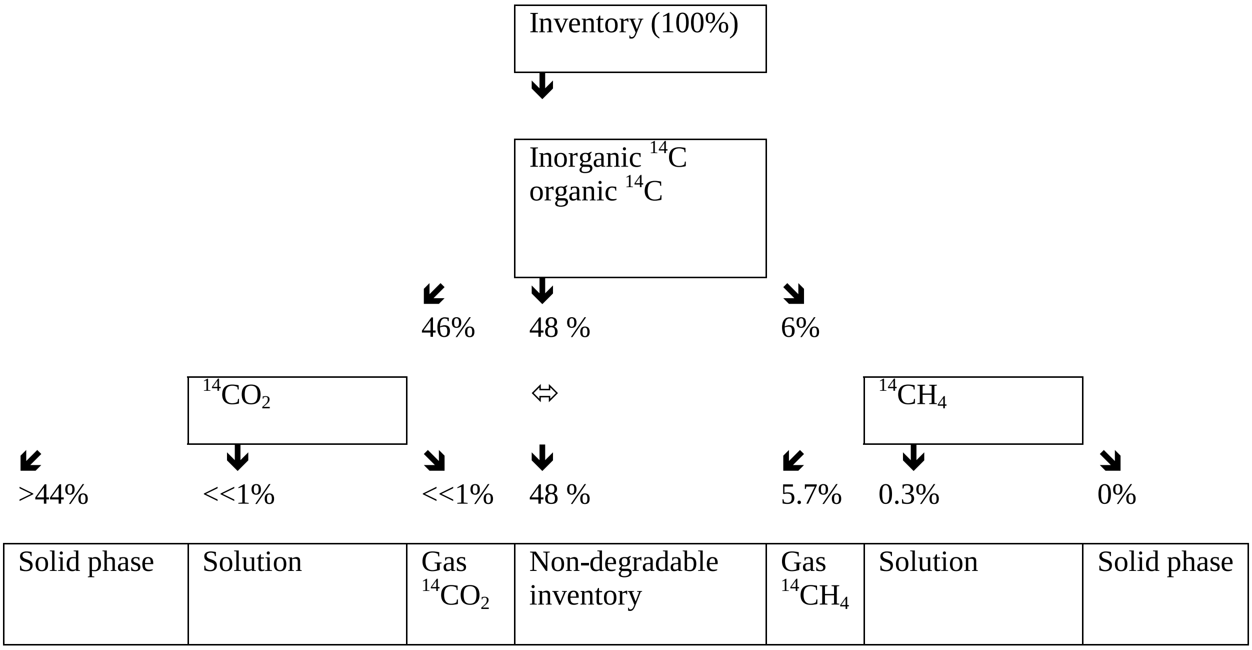
Figure 2 depicts an estimated relative distribution of C-14 in all phases as a result of our approach for one example with a specific closure concept, which considers transport and other effects.

The underlying assumptions are as follows:

1. C-14 activity is homogeneously distributed and mainly bound to organic materials. Occurrences of C-14 in other phases such as carbides or inorganic compounds with different release patterns in time are neglected.
2. A maximum generation of inorganic carbon by oxidation takes place.
3. A realistic gas generation rate is applied to CH_4 which includes methanogenesis
4. Additional dilution by isotope exchange of CH_4 is neglected.
5. Oxidation of CH_4 on transport and subsequent precipitation is neglected.
6. Equilibration of carbonates (solid phase, gas phase, brine) is instantaneous.
7. C-14-containing intermediates are neglected.
8. Released C-14 activity before mine closure is neglected.

The assessment shows that less than 1% of the C-14 activity is released on the fluid pathway. The gas pathway releases approximately 5.7% to the overlying rock and can contribute to the potential radiation exposure in the biosphere.

Figure 2. **Distribution of C-14 in different compartments**



Conclusions

An approach that considers processes by more realistic modelling will lower the potential radiation exposure significantly.

A greater reduction can be achieved when the modelling accounts for:

- C-14 which is not exclusively bound to organic matter.
- C-14 which occurs in all phases.
- a fraction of C-14 which was released before closure of the final repository.
- C-14 in carbonates which are mainly precipitated and remain bound to the solid phase.
- geochemical processes on the transport of brine which will lower the fraction of C-14 released to the overlying rock.

A simplified, conservative evaluation of the source term of C-14, its transport and its potential radiation exposure in the biosphere seems inadequate since it may lead to a grossly overestimated potential radiation exposure.

A detailed evaluation and assessment of processes is necessary to obtain a realistic estimate of the potential radiation exposure for the long term safety assessment of a final radioactive waste repository in order to comply with limits of radiation protection and have a sufficiently large safety margin.

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PRE-CLOSURE SAFETY ANALYSIS FOR SEISMICALLY INITIATED EVENT SEQUENCES

M.J. Shah
U. S. Nuclear Regulatory Commission, USA

Abstract

This paper describes a methodology to determine probabilities of occurrence of seismically initiated event sequences, considering event occurrence probabilities and performance of structures, systems, and components (SSCs). The methodology can be used to assess safety of the preclosure facility for the seismic hazard at the proposed geologic repository at Yucca Mountain, Nevada, and to demonstrate compliance with the risk-informed, performance-based regulations in the U.S. Code of Federal Regulations, Title 10, Part 63. The probability of occurrence of an event sequence leading to an SSC failure is determined by convolution of the seismic hazard curve with the conditional failure probabilities (i.e. fragility) of the SSCs. The methodology is illustrated using examples of potential event sequences. The methodology described in the paper shows how the safety of a facility during a seismic event can be determined using the performance-based regulations. The scope of the paper is limited to estimating the probabilities of occurrence of potential event sequences leading to failure of SSCs and potential release of radioactivity; it does not discuss dose or risk estimates.

Introduction

The U.S. Nuclear Regulatory Commission's (NRC's) mission is to license and regulate the Nation's use of byproduct, source, and special nuclear materials, to ensure adequate protection of public health and safety, promote the common defense and security, and protect the environment. NRC's regulations are in the U.S. Code of Federal Regulations (CFR), Title 10, Chapter 1 and address nuclear safety throughout the lifetime of the facility. These regulations include requirements to prevent release of radioactive materials in the event of man-made hazards (e.g. equipment failures, human errors, or aircraft crashes) and natural hazards (e.g. earthquake, tornado, and flood). In contrast to a design-based deterministic approach for regulating nuclear facilities, the recent trend has been to use risk-informed and performance-based regulations to demonstrate nuclear safety, consistent with the Commission's Policy Statement (Refs. 1, 2).

Regulations in the U.S. Code of Federal Regulations, Title 10, Part 63 (10 CFR Part 63) (Ref. 3), for the proposed high-level nuclear waste disposal facility at Yucca Mountain (YM), Nevada, USA, are risk-informed and performance-based. These regulations require explicit demonstration of the performance of structures, systems, and components (SSCs) that are relied on to demonstrate nuclear safety, defined as important to safety (ITS). The discussion in this paper is limited to a methodology for demonstrating the performance of SSCs ITS during a seismic event, for compliance with 10 CFR Part 63. The scope of the paper is limited to estimating the probabilities of occurrence of potential event sequences leading to failure of SSCs and potential release of radioactivity; it does not discuss dose nor risk estimates.

It should be noted that the methodology discussed in this paper is for the pre-closure safety analysis of the proposed geologic repository at YM. However, the methodology is general, and can be applied to evaluate the performance of engineered barriers during the longer time frames of the post-closure period of the geologic repository, if SSC conditional failure probabilities (i.e. fragility curves) are developed, based on considering the effects of the longer time periods on material capacities.

Discussion

Regulations in 10 CFR Part 63 are performance-based. Instead of specifying design requirements, 10 CFR 63.111 specifies performance-based standards for the geological repository operations area (GROA) as radiological dose limits, to the public, for Category 1 and 2 event sequences. Category 1 event sequences are those that are expected to occur one or more times before permanent closure of the GROA. Category 2 event sequences are those other event sequences that have at least one chance in 10 000 of occurring before permanent closure of the GROA. Event sequences with the probability of occurrence less than that of a Category 2 event sequence are screened out.

To meet the performance objectives of 10 CFR 63.111 for seismic hazard, the preclosure safety analysis must include: (1) a systematic examination of the site; (2) characterisation of the seismic hazard; (3) resulting event sequences; and (4) potential radiological exposures to the public. Based on the review of these event sequences, and the potential release of radioactive material and estimated doses, SSCs ITS must be evaluated to demonstrate their ability to perform intended safety functions under seismic loads.

Seismic Performance Demonstration

The probability of occurrence of seismic initiating events and the failure probabilities of SSCs need to be considered to demonstrate that SSCs ITS will perform their intended safety functions. The probability of occurrence of an event sequence leading to an SSC ITS failure, is determined by convolution of the mean seismic hazard curve with the mean conditional failure probabilities (i.e. fragility) of the SSCs ITS (Ref. 4). The mean fragility curve for an SSC ITS may be estimated using: (1) probability density functions for controlling parameters in a Monte Carlo analysis; (2) the simplified methods outlined in Section 4 of Electric Power Research Institute, TR-103959 (Ref. 5); or (3) other methods that capture the appropriate variability and uncertainty in parameters used to estimate the capacity of the SSCs ITS to withstand seismic events.

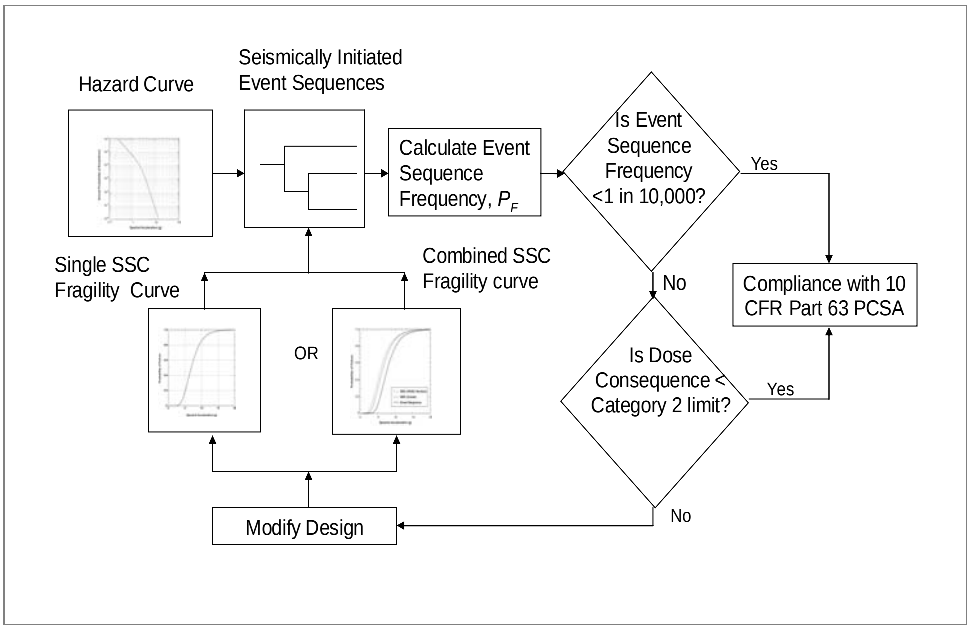
The methodology described herein is based on evaluating event sequences for seismically initiated events and identifying SSCs ITS for seismic performance evaluation. The first step in estimating the probability of occurrence of seismic event sequences is to assess the seismic performance of the individual SSC ITS. For example, to obtain the mean fragility curve of the individual SSC ITS, the median capacity ($C_{50\%}$) and the composite logarithmic standard deviation (β) should be estimated using transparent technical bases. Failure criteria used for estimating the fragility curves should be consistent with the SSCs ITS functional requirements. The mean annual failure probability of the individual SSCs ITS can then be obtained by convolving the mean seismic hazard curve at the site, and the mean fragility curve.

If the annual probability of failure values of individual SSCs ITS for seismically initiated event sequences, estimated using the methodology discussed above, is less than 1 in 10 000 during the preclosure period, as defined in 10 CFR 63.2 for Category 2 event sequences, the SSC ITS is considered to perform its intended safety function and meets 10 CFR 63.111. If, however, the annual probability of failure of the individual SSCs ITS for seismically initiated event sequences is greater than, or equal to, 1 in 10 000, during the preclosure period, DOE may demonstrate compliance with

10 CFR 63.111 by showing that the annual probability of occurrence of each of the seismic event sequences containing multiple SSCs ITS in an event sequence is less than 1 in 10 000 during the preclosure period. Alternatively, the dose consequence to the public at the site boundary from the event sequence can be calculated to show that the dose is less than the dose limits in 10 CFR 63.111(b)(2). It should be noted that Part 63 regulations require consideration of a single seismically initiated event sequence for meeting the dose performance limit, in contrast to consideration of all seismically initiated event sequences to determine the dose in a probabilistic risk analysis of a nuclear reactor facility.

The methodology described above is shown in a flowchart, in Figure 1.

Figure 1. **Methodology for evaluation of seismically initiated event sequences**



SSC – structure, system, or component
PCSA – preclosure safety analysis

Conclusion

Performance of SSCs relied on to ensure nuclear safety during a seismic event can be determined using a methodology, as described in this paper, to demonstrate compliance with the risk-informed performance-based regulations of Part 63. The methodology is based on the convolution of the seismic hazard curve and the SSC conditional failure probabilities or fragility curves.

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SAFETY ASSESSMENTS OF ULTIMATE ISOLATION OF RADIOACTIVE WASTE IN DEEP GEOLOGICAL FORMATIONS IN RUSSIA: OBJECTIVES, PROBLEMS, PROSPECTS

T.A. Gupalo
FSUE VNIPIpromtekhnologii, Rosatom, Russian Federation

Since the 1970s the nuclear countries engaged in radioactive waste management have been developing the criteria of suitability of geological conditions for waste disposition. A large number of guides on suitable site selection based on geoscientific reasons and issued by the appropriate organisations have been developed and applied both nation-wide and internationally. Many of them have been formulated following the recommendations for the properties of the basic isolation barrier, i.e. rock massif that prevents long-term radionuclide release from the repository into the biosphere.

When defining criteria for the best geological sites assessment many Russian authors prefer to use the system approach based on the sum of ratings, arranging various factors describing sites from geological, tectonic, seismic, hydro-geological, mining and engineering and other positions. But the list of qualitative factors differs, assignment of ratings is a subjective parameter, and only objective characteristics based on measured rock parameters, structure and waste amount are required for expert estimations. Recent reviews and analyses of the domestic and foreign literature concerning criteria of site selection have shown the distinctive and common signs of qualitative estimates for site prioritisation estimating the role of various factors. The authors divided the importance of these factors into four types of criteria that help comparative quality assessment of performance for the sites with the best isolating properties in various adverse scenarios.

The analysis of recent research into the problem of site suitability justification for HLW or SNF underground isolation has shown that all approaches to site selection under consideration demonstrate common disadvantages:

- they are in the form of qualitative recommendations;
- disregard quantity and composition of waste assumed for isolation on the selected site;
- lack a common generalising criterion that combines various factors of multi-barrier protection state variation which determine the safety of radioactive waste and SNF.

An optimum decision on the choice of a site for waste disposal is determined from the ultimate goal of RW disposal safety analysis, i.e. minimisation of risk of environmental contamination by radionuclides.

The last publication generalises the main principles of RW geological disposal realised in the majority of European Union member countries, and offers rock qualitative characteristics such as low permeability, high thermal conductivity, favorable geological properties, weak underground waters movement etc. as the limiting values for the criteria of rock properties and sites characterisation.

The risk of the situation when underground waters with maximum permissible concentration of radionuclides achieve the border of active water exchange area is offered as a more rigid criterion.

Most of the international projects use the expected individual dose for population in repository location area with the account of potentially possible scenarios of radionuclide release from repository as a safety indicator. The values of a safety criterion for HLW disposal used nowadays are reduced to the values 10^{-4}-10^{-6} 1/year, in many countries the period of safety analysis is accepted equal to 10 000.

"Accident" is another widely used concept estimating safety of nuclear power plants. The system of classification of accidents and abnormal situations (incidents) at nuclear power plants on the international scale INES is introduced for efficient estimation of events and international notification.

However, the concept of accident on the INES scale is inapplicable for all waste disposal techniques, since even in case of radionuclide release from the repository in an active water exchange zone with concentration exceeding the allowable safety norm, the level of this event in terms of its possible consequences after dilution can be only between the 1st and 2nd levels of INES scale, i.e. no obvious exposure for the personnel, population and contamination of the territory will be observed (Figure1).

The approach to selection of geological sites for underground isolation of solid and solidified radioactive waste is offered on the basis of risk calculations of radionuclide release from underground facilities. The specific feature of the given approach is that probabilities of occurrence of abnormal situations and events included in emergency sequences are estimated on the basis of physical processes parameters determined in natural conditions in view of their changes during the required period of waste isolation owing to various natural and technogenic factors.

The approach offered allows the safety of waste disposal to be efficiently estimated proceeding from the actual properties of the site and waste at any phases of design, construction and operation of underground storage facility or repository.

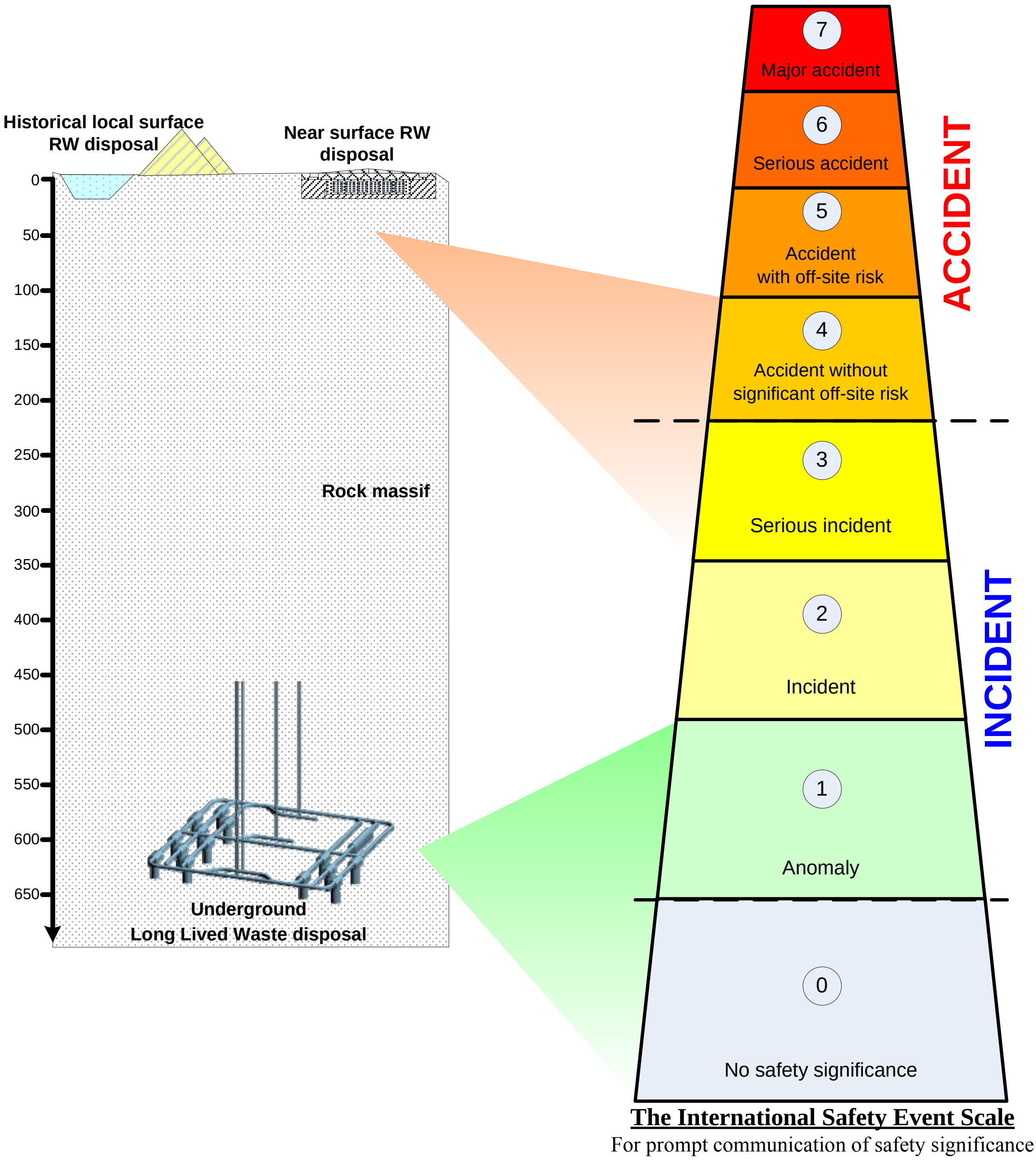
Publications on safety analysis of HLW and SNF final disposal use an annual total individual dose which can be received by the population inhabiting the area of disposal location as the basic criterion. Mathematical modeling of the following processes has been carried out for the calculation of forecast estimations: change of waste radionuclide content in time as a result of nuclear transformations; leaching radionuclides from matrixes as a result of interaction with underground waters after destruction of engineered barriers system; migration of radionuclide solutions through the engineered barriers system and further on along the fracturing zones of the rock massif.

The so-called "realistic" and "conservative" estimations of such values as rock mass fracturing and speed of underground waters movement in various zones, solubility of radionuclides in underground waters, factors of radionuclides distribution at interaction of contaminated underground waters with host rocks and materials of engineered barriers, duration of through corrosion of containers, etc. are used for calculations.

Food chains, through which radionuclides penetrate a human body, are analysed similar to the techniques developed for safety estimation of nuclear power plants. The "risk" indicator is used for estimation of a risk of possible death includes the estimated additional individual dose owing to radionuclide release to the sphere of life activity.

However, the expected individual dose for forecasting period of radionuclides preservation of their hazard (the hundreds of thousands years). Therefore another criterion is necessary for safety assessment of repositories, i.e. an indicator which can be evaluated on a real time basis and predicted for a long-term period of time in a massif at any phase of selection of a place for repository operation and closure.

Figure 1. **Assessment of consequences of routine operation and incidents at isolation facilities with RW of various types by the INES scale**



The new approach to a choice of geological sites was developed under the author's guidance based on the probabilistic safety analysis of multi-barrier systems of RW and SNF long-term underground storage and geological disposal. Main principles of the approach are as follows:

- A safety criterion is the risk of radionuclides with ecologically hazardous concentration in underground waters achieving the border of active water exchange area; no intake of radionuclides into the sphere of life activity with activity in water higher than that admitted by the sanitary norms is allowed for risk calculation; the allowable risk value is accepted as 10^{-6} 1/year;
- Abnormal situations which can arise during the given period of storage are analysed for underground facilities; the risk of radionuclide release is estimated in view of probability of occurrence of each abnormal situation and its possible consequences;
- The risk of release for HLW disposals is determined for the whole period of potential hazard of radionuclides composing the waste; the list of most ecologically dangerous radionuclides is defined based on the calculations;
- In order to increase reliability of safety assessments, the total annual individual dose from water intake is estimated for radionuclides with the risk exceeding 10^{-6} 1/year.

Figure 2 presents the block diagram for calculations on substantiation of sites suitability for HLW and SNF long-term and ultimate isolation underground facilities, and also for optimising design decisions by the complex criterion "risk - expenses".

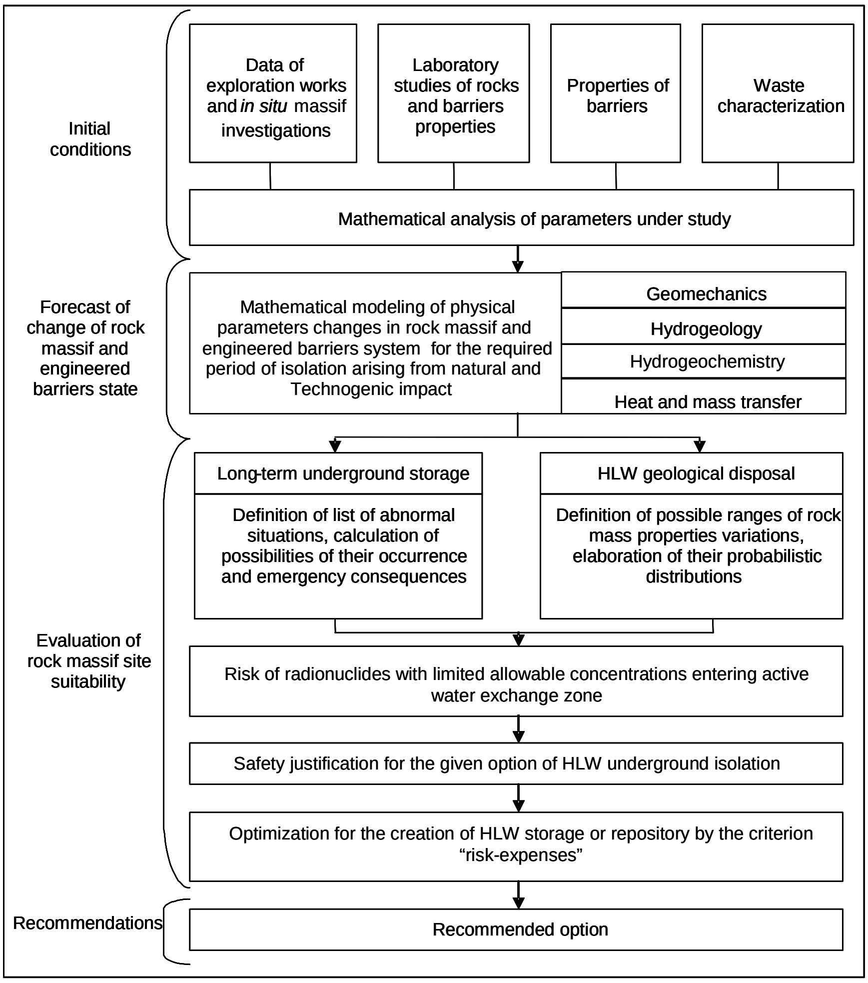
In justification of the possibility to create SNF, long lived HLW, ILW repository within the limits of a specific site, the estimation of ecological safety is carried out with the account of potentially probable abnormal situations when radionuclides could find their way into the environment.

All physical processes in a rock massif are interrelated, and a change of stressed-strained state in the near-contour zone rock can lead to opening of available cracks, appearance of water-conducting channels, water seepage in a drift and contacting with the stored materials followed by leaching and radionuclides ingress into underground waters.

The list of natural and technogenic factors, which can lead to a significant change of isolation properties of multi-barrier protection and occurrence of abnormal situations, is made depending on rock massif performance and design features of underground facility. There is also an international list of FEP's related to the natural phenomena, the most significant of which are structural highs and subsidence, activation and formation of new faults, mineralisation and weathering of fissures, hydro-geological changes, etc. Technogenic effect is assessed on the basis of estimations of rock massif state change owing to the construction of underground facility, influence of waste, and deliberate destructive activity inside the facility.

The procedure of probability estimations of abnormal situation occurrence of natural and man-induced character such as earthquakes, geodynamic movements, change of fracturing, drifts flooding, etc. significant for investigated facilities are considered.

Figure 2. **Block diagram for quantitative justification method of rock massif suitability for HLW long-term underground isolation**



For HLW and SNF underground repositories with the given period of storage for every radionuclide the risk of achievement by aureole of contaminated underground waters with the concentration above the allowable of active water exchange area is estimated, the distance up to this area along fracturing zones being equal. The received values are compared to the limiting value 10^{-6} 1/year.

Besides, the probabilities of radionuclide release into active water exchange area at abnormal situations or in their absence during the entire period of isolation are calculated as a multiplication of conditional probabilities of events included in corresponding emergency sequences.

The probabilities of occurrence of abnormal situations and conditional probabilities of radionuclides overcoming each barriers of a multi-barrier system depend on the environment of the area, content and volumes of HLW, rock mass and engineered barriers performances with the account of technogenic effects of repository.

The probabilities of occurrence of potentially possible abnormal situations and conditional probabilities of events included in corresponding emergency sequences are determined on the basis of analysis and forecasting of environmental conditions of the district, structure and performances of the rock massif at the site of repository location, strength properties of underground facility, engineered barriers properties, hydro-geological and physical and chemical characteristics of the rock massif under study, composition of waste and underground waters. Moreover mathematical modeling of stressed-strained state of near-contour zones of rock massif and calculation of strength characteristics of "rock massif-support" system in view of possible dynamic impacts of various intensity and results of in situ measurements in rock massif and support are carried out.

Figure 3 presents a general calculation scheme and examples of definition of probability of abnormal situations occurrence based on the results of in situ investigations in rock massif ("Geodynamic shift", "Drift local failure"), mathematical modeling of stability of underground facilities under conditions of slowly changing ("Geodynamic shift" and "Drift local failure") and dynamic ("Earthquake") impacts, analysis of statistical data of the area ("Earthquake").

Main problems that hinder the safety analyses which are faced by researchers worldwide are the obtaining of input data which describe processes of radionuclide contamination migration in the geological environment (migration characteristics) and forecasting the dynamics of changes in host rock parameter values.

To determine migration parameter values, the field studies are carried out in Russia on behavior of radionuclide contamination having a complex composition in rock mass; their results can be used by international researchers as input data for the probabilistic safety analysis.

Figure 3. **Calculation scheme of abnormal situation occurrence probabilities for HLW underground isolation facilities**

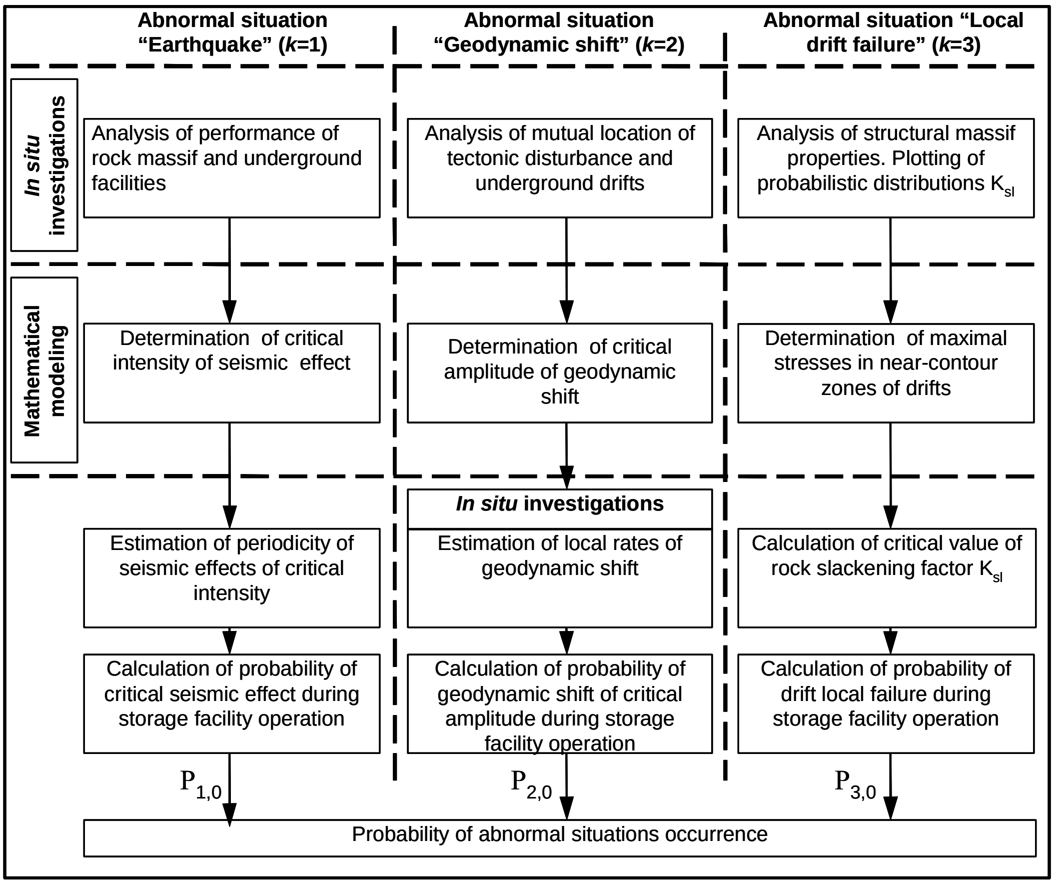
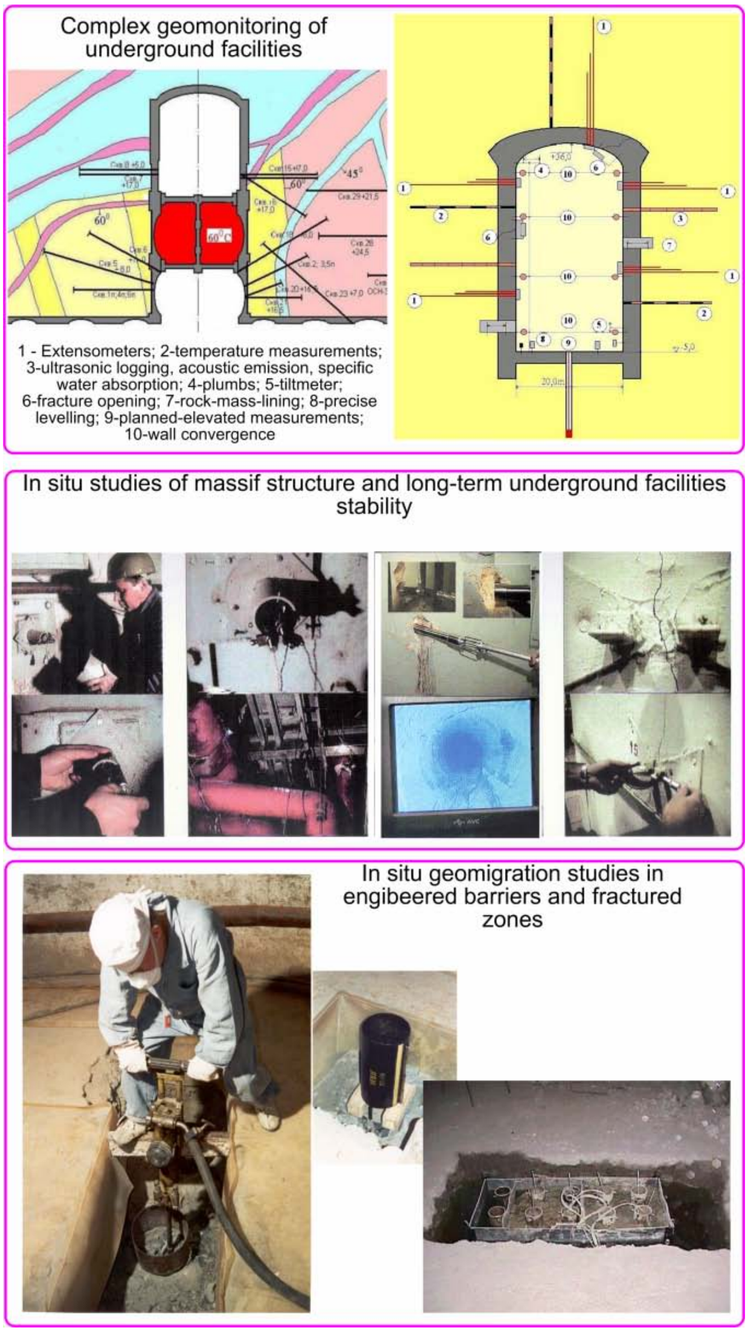


Figure 4 represents the basic researches carried out in rock massifs to define the values important for safety parameters.

Figure 4. **Basic research performed in the underground facilities for evaluation of the parameters determining the final isolation safety**



The experience accumulated in Russia in operation of underground facilities which have been producing heat impacts to the host rock for decades that are similar to impacts to be produced by the RW ultimate isolation facility, can be used to solve the issue of forecasting parameter changes of the geological environment hosting the RW repository.

DEVELOPMENT OF THE SAFETY CASE OF GEOLOGIC REPOSITORY IN KOREA

C-H. Kang, Y-M. Lee, Y-S. Hwang
Korea Atomic Energy Research Institute, Korea

Abstract

In Korea, its national energy supply is dependent on the peaceful and safe utilisation of nuclear energy. Since April of 1978, Korea has strongly relied on nuclear energy for its electricity generation. In Korea, twenty nuclear reactors are in operation, and four are under construction. By 2017 twenty-eight nuclear reactors will be in operation with a nuclear share of 47.6%, unless old reactors are to be retired. The Korean Government's policy on nuclear energy can be summarised into two principles, promotion of the peaceful uses of nuclear energy and securing its safety.

The Korea Atomic Energy Research Institute (KAERI) launched a three-step 10-year R&D programme in 1997 to develop a reference geologic repository system for HLW by 2006. A preliminary reference repository concept for spent PWR and CANDU fuel was proposed. In parallel, the overall system performance assessment code for a probabilistic safety assessment has been developed. To accomplish a TSPA, a systematic development of the FEP list was pursued. Based on the KAERI FEP Encyclopedia, reference and alternative scenarios were developed. By applying the TSPA code, a preliminary safety assessment was performed by using a generic data set. Results showed that the current disposal concept proposed by KAERI satisfies the safety criteria for the given radionuclide release scenarios.

Introduction

The Korean Atomic Energy Commission provided clear milestones to construct two key facilities, the LILW repository by 2008 and a centralised storage for spent fuel by 2016. Even though a proper scientific management of radioactive waste has been proven for a LILW disposal and an interim storage of spent nuclear fuel and the introduction of a permanent disposal of HLW is envisaged to be feasible, still the public perception on these issues are not so supportive. To overcome the issues of the safety of a radioactive waste management is not a simple scientific problem. It is the combination of the state of the art science and technology with the psychology of the general and local public. Long term safe management of spent nuclear fuel is very important not only to store spent fuel over a long period of time but also to potentially recycle it under full observance of the nuclear non-proliferation duties.

Korea Atomic Energy Research Institute (KAERI) has been working on the development of a permanent disposal of HLW and its total system performance assessment since 1997. The current research and development activities are focused on a preliminary conceptual design study to set up the Korean reference repository system (KRS) and a total system performance assessment for the KRS.

Repository System Development

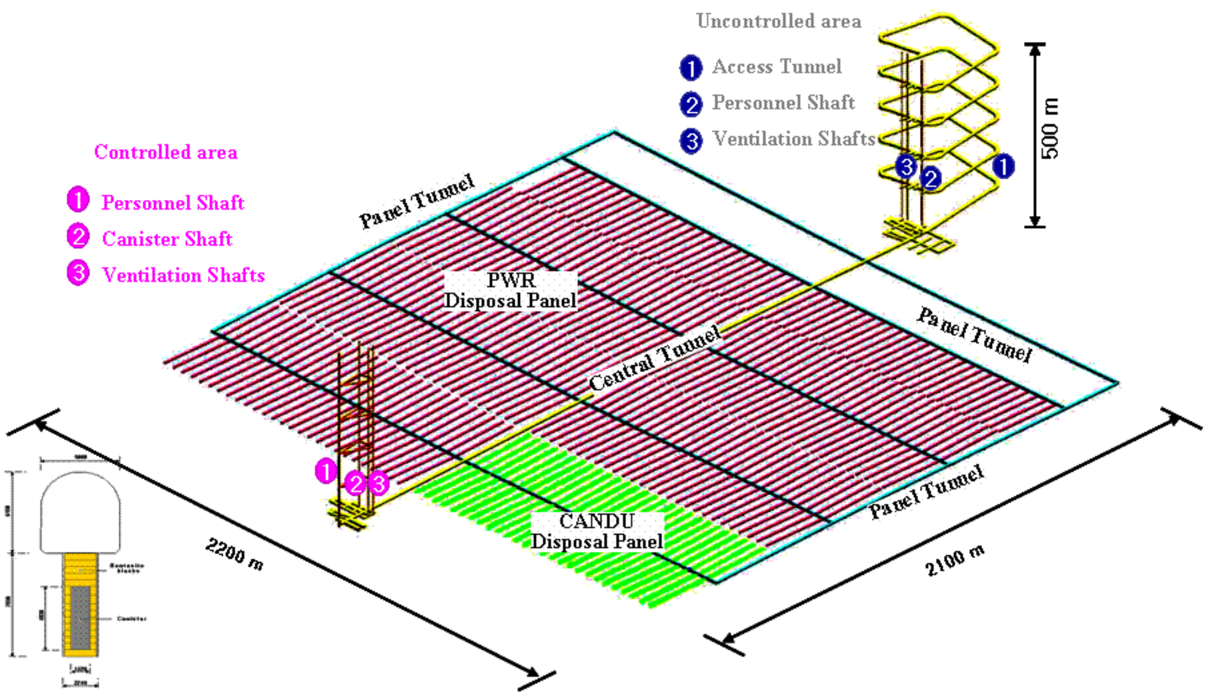
The packaged spent fuel encapsulated in a corrosion resistant container is to be disposed of in a mined underground facility, located at about 500 m depth in a crystalline rock mass [1,2]. No site for the underground repository has been specified in Korea, but a generic site with a granite rock has been considered. The waste packages that contain spent fuel are placed in boreholes drilled into the floor of disposal drifts.

The reference PWR spent fuel has an average burn-up of 45 000 MWd/tHM (initial enrichment of 4.0 wt%) and the fuel dimensions are 21.4cm × 21.4cm × 453cm (length). The reference CANDU fuel has an average burnup of 7 500 MWd/tHM and the fuel dimensions are 10 cm (diameter) × 49.5cm (length). Because of the significantly different properties of both fuel types, the reference container encapsulates the spent PWR and CANDU fuels separately [3]. However, the overall sizes and component materials of the containers for both spent fuel types are proposed to be identical to simplify the encapsulation and handling processes in the repository. The dimensions of the container were determined from a mechanical structural analysis under the expected mechanical loads in underground repository conditions [4]. Bentonite is under consideration as the buffer material because of its low permeability, high sorption capacity, self-sealing characteristics, and durability.

The disposal area consists of 8 disposal panels including a future expansion. Based on a 40 m emplacement tunnel spacing, the disposal tunnels for the PWR and CANDU fuels are 323 and 54 tunnels, respectively (see Figure 1). Each emplacement tunnel is 254 m long which includes about a 10 m end-standoff and about a 17 m standoff at the entrance for the emplacement works.

Alternative disposal concepts being considered in the study include multiple level waste emplacements with a shaft access. These, in turn, have a potential impact on a disposal room layout, ventilation, construction sequence, etc. Each concept has construction-related features which have advantages and disadvantages.

Figured 1. **The Korean Reference Repository System**



Total System Performance Assessment

The major activities of the performance assessment [5] to see whether the KRS concept can satisfy the regulatory guidelines are to develop the tools and database for a scenario development, to develop the needed computational tools and database for the TSPA, and to apply the TSPA tools and database to assess the safety of the disposal concepts. To meet these demands KAERI has accomplished the following:

1. Development of the FEAS (FEP to Assessment through Scenarios) package which includes the KAERI FEP encyclopedia, rock engineering system (RES) matrices, assessment methods, etc.
2. Data collection activities and development of PAID (Performance Assessment Input Database) by using two folder systems based on material and parameter names in association with barriers.
3. Assessment of the groundwater flow in a fractured porous media by using the CONNECTFLOW.
4. Understanding the Korean biosphere by focusing on three different critical groups, farmers and fresh and marine water fishing groups.
5. Development and verification of the MDPSA(Multi Dimensional Probabilistic Safety Assessment) and MASCOT-K to simulate the transfer and transport of a radionuclide from a failed waste container to the biosphere via a bentonite buffer and fractured porous media;
6. Performing assessments each year to estimate the annual individual.
7. Development of a web based information system including a quality assurance system and combining it with other components such as FEAS, PAID, MASCOT-K and MDPSA in addition to the new documentation system [6].

The preliminary TSPA is pursued for reference concepts in a coastal area. In this study, the main focus is to assess the safety for different sets of scenarios. The first set of scenarios is the reference scenarios in Figure 2. In these scenarios a waste container fails after a 1 000 year design lifetime. The two dissolution mechanisms are a congruent and instantaneous release from a uranium oxide matrix. The released radionuclides pass through the bentonite buffer and reach a fracture network. The second group of scenarios is the failure of engineered barriers. Here we assume some initial failures of the waste containers, a malfunctioning of the buffer, and the impact of an excavated disturbed zone. The third group of scenarios is the natural disruptive scenarios. In this case we consider the advent of an ice age, the potential reactivation of a fault, and a sudden change of the groundwater flow rates.

All the analyses indicate that the predicted annual individual doses are below the 0.1 mSv/yr dose target currently prescribed for the LLW disposal case (see Figures 3 and 4).

Figure 2. **The Concept of the Reference Scenarios**

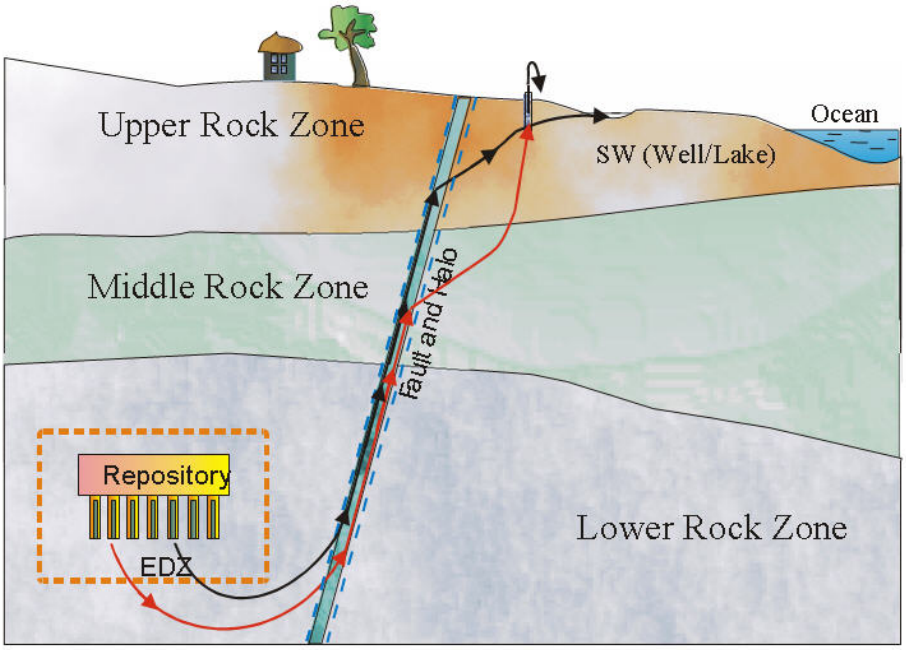


Figure 3. **Annual Individual Dose Rate for a reference scenario**

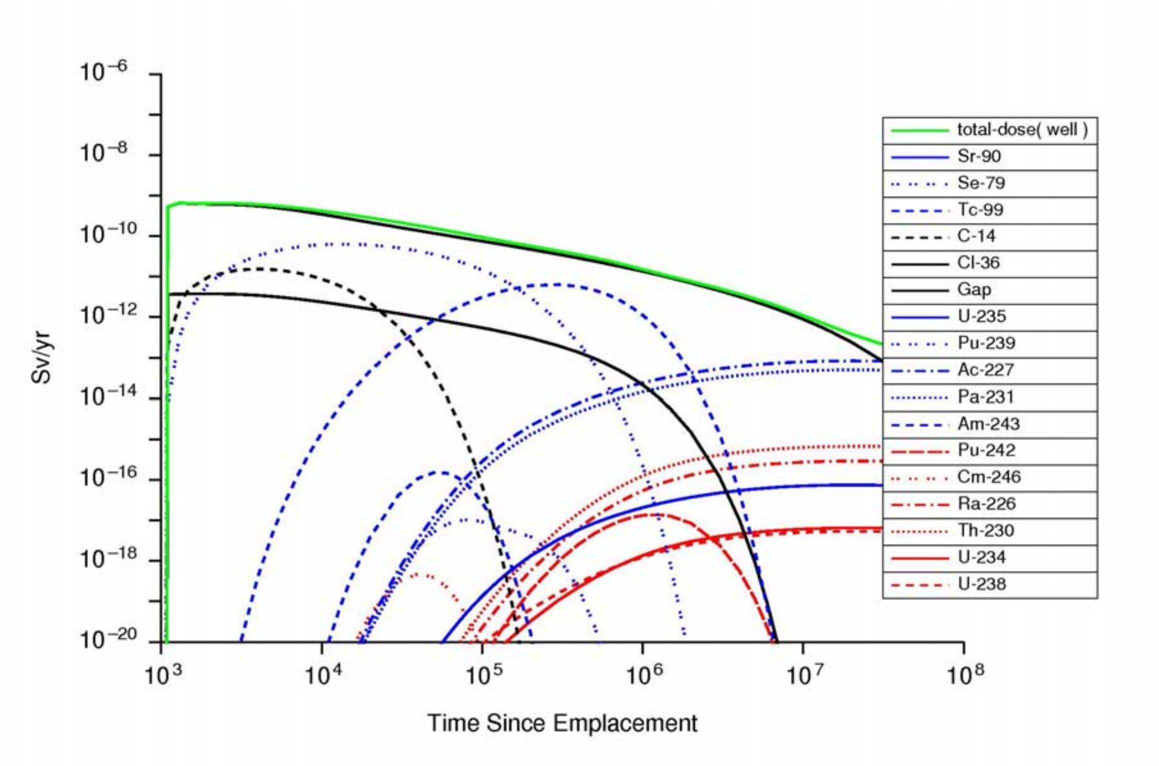
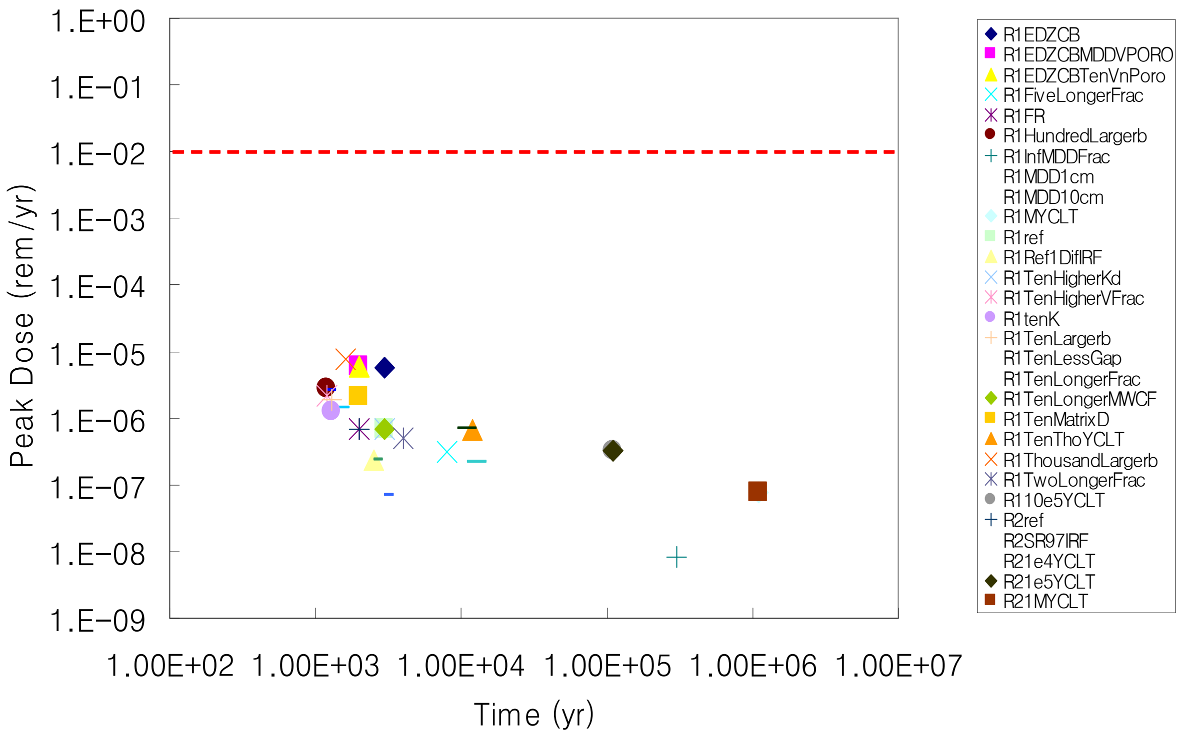


Figure 4. **Annual Individual Dose Rate for different scenarios**



Summary

A preliminary reference repository concept for spent PWR and CANDU fuel was proposed. A site for the underground repository has not been specified as yet, but a generic site with a granite rock was suggested for this study. The waste packages will be vertically placed in boreholes drilled into the floor of the deposition tunnels or horizontally placed in the deposition tunnels located about 500 m below the surface in a crystalline rock mass. In parallel, the overall system performance assessment code for a probabilistic safety assessment has been developed. To accomplish the TSPA, a systematic development of the FEP list was pursued. Based on the KAERI FEP Encyclopedia, reference and alternative scenarios were developed. By applying the TSPA code, a preliminary safety assessment was performed by using a generic data set. Results showed that the current disposal concept proposed by KAERI satisfies the safety criteria for the given radionuclide release scenarios. In the future, a detailed TSPA will be performed by using the Korean geologic data and a more detailed disposal concept.

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Appendix F

LIST OF PARTICIPANTS

BELGIUM	Dr. Peter DE PRETER ONDRAF/NIRAS Avenue des Arts, 14 B-1210 Brussels
	Mrs. Ann DIERCKX ONDRAF/NIRAS Avenue des Arts, 14 1210 Brussels
	Dr. Jan MARIVOET SCK•CEN Boeretang, 200 2400 Mol
	M. Vincent NYS Association Vinçotte Nuclear (AVN) 148, rue Walcourt 1070 Bruxelles
	Mr. Olivier SMIDTS Association Vinçotte Nuclear rue Walcourt, 148 1070 Bruxelles
	Professor Eric VAN HOVE University of Antwerp Nieuwe Baan 2 3730 Hoeselt
	Mrs. Eef WEETJENS SCK•CEN Boeretang 200 2400 Mol
CANADA	Mr. Richard FERCH Expert to NEA 1267 Marygrove Circle K2C 2E1 Ottawa
	Mr. Peter FLAVELLE Canadian Nuclear Safety Commission PO Box 1046Station B280 Slater Street K1P 5S9 Ottawa
	Mrs. Theodora KEMPE OPG 700 University Avenue M5G 1X6 Toronto,
	Dr. Helen LEUNG Ontario Power Generation 700 University Avenue (H17) M5G 1X6 Toronto
CZECH REPUBLIC	Mrs. Sona KONOPASKOVA RAWRA Dlazdena, 6 110 00 Praha 1

FINLAND

Dr. Kari KOSKINEN
Posiva Oy
27160 Olkiluoto

Mrs. Nuria MARCOS
Saanio & Riekkola Oy
Laulukuja, 4
00420 Helsinki

Mr. Mikko NYKYRI
Safram Oy
Metsätie 24
02300 Espoo

Mrs. Barbara PASTINA
Saanio & Riekkola Oy
Laulukuja 4
FI-00420 Helsinki

FRANCE

M. Philippe BODENEZ
Autorité de Sûreté Nucleaire (ASN)
10, Route du Panorama – BP83
92266 Fontenay-aux-Roses Cedex

M. Jean-Yves BOISSON
DEI/SARG
IRSN
BP 17
92262 Fontenay-aux-Roses Cedex

Mr. Marc BUTEZ
CEA/SACLAY
DEN/DPA
BAT. 121
91191 Gif-sur-Yvette Cedex

Mr. Bruno CAHEN
Andra
1-7, rue Jean Monnet
92298 Châtenay-Malabry Cedex

Mr. Bernard FAUCHER
Andra
1-7, rue Jean Monnet
92298 Châtenay-Malabry Cedex

M. Eric GAUCHER
EPI/DES Déchets et Stockage
BRGM
B.P 6009
45060 Orléans cedex 2 Orléans

Mme Lise GRIFFAULT
Andra
1-7, rue Jean Monnet
92298 Châtenay-Malabry Cedex

Mr. Thibaud LABALETTE
Andra
1-7, rue Jean Monnet
92298 Châtenay-Malabry Cedex

FRANCE (Cont.)	Mr. Patrick LANDAIS Andra 1-7, rue Jean Monnet 92298 Châtenay-Malabry Cedex
	M. Gerald OUZOUNIAN Andra 1-7, rue Jean Monnet 92298 Châtenay-Malabry Cedex
	Mrs. Delphine PELLEGRINI DSU/SSD/BECS IRSN BP 17 92262 Fontenay-aux-Roses Cedex
	Mr. Frédéric PLAS Andra 1-7 rue Jean Monnet 92298 Châtenay-Malabry Cedex
	M. Christophe SERRES DSU/SSD IRSN BP 17 92262 Fontenay-aux-Roses Cedex
	Dr. Annie SUGIER DSDRE/MPP Institut de Radioprotection et de Sûreté Nucléaire (IRSN) 77 - 83 avenue du Général de Gaulle 92140 Clamart
	Mme Sylvie VOINIS Andra 1-7 rue Jean Monnet 92298 Châtenay-Malabry Cedex
GERMANY	Professor Wernt BREWITZ Gesellschaft für Anlagen und Reaktorsicherheit (GRS)mbH Theodor-Heuss-Str. 4 38122 Braunschweig
	Dr. Klaus-Juergen RÖHLIG Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH Schwertnergasse 1 50667 Köln
	Mr. Bruno BALTES Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH Schwertnergasse 1 50667 Köln
	Dr. Dirk-Alexander BECKER Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH Theodor-Heuss-Str. 4 38122 Braunschweig

GERMANY (Cont.)

Dr. Enrique BIURRUN
DEB Technology GmbH
Eschenstr.55
31224 Peine
Dr. Guido BRACKE
ISTec GmbH
Schwertnergasse 1
50667 Köln

Mr. Gunnar BUCKAU
Forschungszentrum Karlsruhe - Insitut für Nukleare Entsorgung (FZK-INE)
P.O. Box 3640
76021 Karlsruhe

Mr. Dieter BUHMANN
Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH
Theoder-Heuss-Str. 4,
38122 Braunschweig

Dr. Klaus FISCHER-APPELT
Gesellschaft für Anlagen und Reaktorsicherneit mbH (GRS)
Schwertnergasse, 1
50667 Köln

Dr. Bernd FÖRSTER
GSF Forschungsbergwerk Asse
Am Walde 2
38319 Remlingen

Mrs. Beate KALLENBACH-HERBERT
Oeko-Institut
Rheinstrasse, 95
64295 Darmstadt

Mr. Juergen KRONE
DBE Technology
Eschenstrasse 55
31224 Peine

Dr. Kerstin KUEHN
BfS
P.O. Box 100149
38201 Salzgitter

Mr. Peter-Jürgen LARUE
Gesellschaft für Anlagen und Reaktorsicherheit mbH (GRS)
Schwertnergasse, 1
50667 Köln

Mr. Karl-Heinz LUX
Institut f. Aufbereitung u. Deponie-technik
TU Clausthal
Erzstraße 20
38678 Clausthal-Zellerfeld

Mr. Jörg MÖNIG
GRS-Braunshweig
Theodor-Heuss-Str. 4
38122 Braunschweig

GERMANY (Cont.)

Mrs. Nina MÜLLER-HOEPPE
DBE/DBE Technology
Eschenstr, 55
31224 Peine

Ingo MULLER-LYDA
Gesellschaft für Anlagen und Reaktorsicherheit mbH (GRS)
Theodor-Heuss-Straße 4
38122 Braunschweig

Mr. Torsten PAPE
LBEG
Stilleweg, 2
30655 Hannover

Mr. Michael SAILER
Oeko-Institut e.V.
Rheinstrasse, 95
65295 Darmstadt

Mr. Hajo SCHNIER
BGR
Stilleweg, 2
30655 Hannover

Mr. Hartmut SCHÜTTE
Lower Saxony Ministry of Environment
Archivstr. 2,
30169 Hanover

Dr. Walter STEININGER
Forschungszentrum Karlsruhe, PTKA-WTE
Hermann-von-Helmholtz-Platz 1
76344 Eggenstein-Leopoldshafen

Mr. Jens VON DEN EICHEN
Procedures under Mining Law, Exploration and Repository Projects
States Authority for Mining, Energy and Geology
An der Marktkirche 9,
38678 Clausthal-Zellerfeld

Mr. Jan Richard WEBER
Bundesanstalt für Geowissenschaften und Rohstoffe
Stilleweg, 2
30655 Hannover

Dr. Juergen WOLLRATH
Bundesamt für Strahlenschutz (BfS)
P.O. Box 10 01 49
Willy-Brandt-Strasse 5
38201 Salzgitter-Lebenstedt

HUNGARY

Mr. Jozsef CSICSAK
MECSEKERC Ltd.
19. Esztergár L. u.,
H-7633 Pécs

Dr. Ferenc FEDOR
MECSEKERC Ltd.
Esztergár Lajos u. 19
H-7633 Pécs

HUNGARY (Cont.)

Mr. Viktor FEURER
Intercomp Ltd
Attila u. 119
H-1047 Budapest,

Mr. Zoltan NAGY
PURAM
Paks Headquarters
PO Box 12
H-7031 Paks

Dr. Istvan SZUCS
Mecsek Ore Environment
Esztergar L. u. 19
7633 Pecs

ITALY

Mr. Alberto TAGLIONI
ENEA
via Anguillarese n.300
00123 Roma

JAPAN

Mr. Shigenobu HIRUSAWA
Research and Development Division
Institute of Applied Energy
Shinbashi Sy Bldg
14-2 Nishi-Shinbashi, 1-chome
Minato-ku
105-0003 Tokyo

Mr. Takashi ICHINOMIYA
Nuclear and Industrial Safety Agency (NISA)
kasumigaseki 1-3-1
chiyodaku
Tokyo

Dr. Katsuhiko ISHIGURO
Nuclear Waste Management Organization of Japan (NUMO)
Mita NN Bldg. 1-23 Shiba4-Chome Minato-ku
108-0014 Tokyo

M. Hiroto KAWAKAMI
Radioactive Waste Evaluation Office
Safety Standards Division
Japan Nuclear Energy Safety Division (JNES)
TOKYO REIT Toranomon Bldg. 7F
3-17-1 Toranomon, Minato-ku
105-0001 Tokyo

Mr. Kazumi KITAYAMA
Nuclear Waste Management Organization of Japan (NUMO)
4-1-23, Shiba, Ninato-ku
108-0014 Tokyo

Mr. Shunichi SUZUKI
Safety Standard Division
Japan Nuclear Energy Safety Organization (JNES)
Radioactive Waste Evaluation Office
Toranomon, Minato-ku
105-0001 Tokyo

JAPAN (Cont.)

Dr. Hiroyuki UMEKI
Geological Isolation Research and Development Directorate
Japan Atomic Energy Agency (JAEA)
2-1-8 UchisaiwaichoChiyoda-ku
100-8577 Tokyo

Mr. Shinichi YAMASAKI
Japan Nuclear Energy Safety Organization (JNES)
Toranomon 3-17-1
Minato-ku
Tokyo

Mr. Hideki YOSHIKAWA
Japan Atomic Energy Agency (JAEA)
Muramatsu 4-33,
Tokai-mura,
319-1194 Ibaraki

KOREA

Dr. Dae-seok BAE
Korea Atomic Energy Research Institute (KAERI)
PO Box 105
Yuseong
305-600 Daejeon city

Mr. Jina JANG
Department of nuclear engineering
Green Nuclear Research Laboratory
Kiheung-gu
Yongin-city

Dr. Chul-Hyung KANG
Korea Atomic Energy Research Institute (KAERI)
P.O. Box 105
Yuseong
305-600 Daejeon

Mr. Sang-Yoon LEE
Department of Nuclear Engineering
519-8 Wooman-dong
442-821 Paldal-Gu

Mr. Seongwoo YANG
Department of Nuclear Engineering, Kyung Hee University
Green Nuclear Research Laboratory
Kiheung-gu
449-701 Yongin-city

Mr. Jaeryong YOO
Department of Nuclear Engineering, Kyung Hee University,
Green Nuclear Research Laboratory (GNRL)
Kiheung-gu
449-701 Yongin-city

LUXEMBOURG

Dr. Patrick MAJERUS
Division de la Radioprotection
Villa Louvigny
Allee Marconi
2120 Luxembourg

NETHERLANDS

Dr. Ewoud VERHOEF
COVRA NV
PO Box 202
4380 AE Vlissingen

RUSSIAN FEDERATION	Professor Tatyana GUPALO VNIPIPremtechnologii 33 kashirskoje shosse 115409 Moscow Russian Federation
	Mr. Alexei GVISHIANI Geophysical centre of Russian Academy of Sciences 3, Molodezhnaya Str. 119296 Moscow
	Mr. Evgeny KAMNEV Institute of Industrial Technology 36, Kashirskoye shosse Moscow
	Mr. Vladislav MOROZOV Geophysical centre, Russian Academy of Sciences 3, Molodezhnaya Str. 119296 Moscow
	Mr. Andrey Ivanonovich RYBALCHENKO VNIPI Promtechnoloy Kashirskoe schosseu 33 115409 Moscow
SPAIN	Mr. Jesus ALONSO ENRESA Calle Emilio Vargas, 7 28043 Madrid
	Mr. José Luis CORMENZANA ENRESA C/Ramírez de Arellano 15 - 1º dcha 28043 Madrid Spain
	Mr. Juan Carlos MAYOR ENRESA Emilio Vargas, 7 Madrid
	Mr. Juan J. MONTESINOS CSN C/Justo Dorado, 11 28040 Madrid
	Dr. Carmen RUÍZ LÓPEZ CSN C/Justo Dorado, 11 28040 Madrid
SWEDEN	Mr. Allan HEDIN Swedish Nuclear Fuel & Waste Management Co. (SKB) Box 5864 102 40 Stockholm
	Dr. Mikael JENSEN SSI 17116 Stockholm

SWEDEN (Cont.)	Mrs. Christina LILJA SKI Klarabergsviadukten 90 106 58 Stockholm
	Mr. Kaj NILSSON Oskarshamn Municipality LKO-projektet Box 706 572 28 Oskarshamn
	Mr. Sören NORRBY KASAM Kungsängsgatan 41C 753 22 Uppsala
	Dr. Eva SIMIC KASAM/Ministry of Sustainable Development Regeringsgatan 30-32 103 33 Stockholm
	Dr. Bo STROMBERG Swedish Nuclear Power Inspectorate (SKI) 10658 Stockholm
	Dr. Öivind TOVERUD Swedish Nuclear Power Inspectorate (SKI) Klarabergsviadukten 90 106 58 Stockholm
	Dr. Anders WIEBERT SSI 171 16 Stockholm
SWITZERLAND	Dr. Matthias NIEMEYER Colenco Power Engineering Ltd. Täfernstrasse 26 5405 Baden
	Dr. Meinert RAHN Swiss Federal Nuclear Safety Inspectorate (HSK) 5232 Villigen-HSK
	Dr. Jürg SCHNEIDER Nagra Hardstrasse, 73 CH-5430 Wettingen
	Dr. Johannes VIGFUSSON Geological Disposal Section Swiss Federal Nuclear Safety Inspectorate (HSK) 5232 Villigen-HSK
	Mr. Piet ZUIDEMA NAGRA Hardstrasse 73 5430 Wettingen
UNITED KINGDOM	Mrs. Lucy BAILEY United Kingdom Nirex Limited Curie AvenueHarwell, Didcot OX11 0RH Oxfordshire

UNITED KINGDOM (Cont.)

Mr. Phil DAVIES
Nuclear Decommissioning Authority
Westlakes Science & Technology Park,
Moor Row
CA24 3HU, Cumbria

Dr. Susan DUERDEN
Nuclear Waste Assessement Team (NRG[N]Penrith)
Environment Agency of England and Wales
Ghyll Mount, Gillan Way
Penrith 40 Business Park
CA11 9BP Penrith

Dr. Alan HOOPER
United Kingdom Nirex Limited
Curie AvenueHarwell, Didcot
OX11 0RH

Mrs. Kelly JONES
Health Protection Agency
Chilton
OX11 0RQ Didcot

Mr. Mike POOLE
United Kingdom Nirex Limited
Curie Avenue, Harwell, Didcot
OX11 0RH

Mr. Neil SMART
Nuclear Decommissioning Authority
Herdus House,
Westlakes Science & Techonology Park, Moor Row,
CA24 3HU Cumbria

Mr. Richard WAITE
Nuclear Decommissioning Authority
Herdus House, Westlakes Science & Technology Park
CA24 3HU Moor Raw, Cumbria

UNITED STATES

Ms. Elizabeth LISANN
Science Advisor for Energy Affairs
Permanent Delegation
12, Avenue Raphaël
75116 Paris
France

Dr. Michael J. APTED
Monitor Scientific
3900 S. Wadsworth
5th Floor, '555
Denver, Colorado 80235

Ms. Bonnie GITLIN
Associate Director
Radiation Protection Division
US Environmental Protection Agency
Mail Code 6608-J
1200 Pennsylvania Ave, N.W.
20460 Washington

UNITED STATES (Cont.)

Mr. David SWANSON
Nye County NWRPO
1210 E. Basin, Suite 6
Pahrump, NV 89060

Dr. Abraham VAN LUIK
Office of Repository Development
US Department of Energy
1551 Hillshire Drive
Las Vegas, NV 89134

Mr. James WINTERLE
Center for Nuclear Waste Regulatory Analysis (CNWRA),
Southwest Research Institute
6220 Culebra Road
Building 189A
San Antonio, TX 78238

Ms. Barbara MERRILL
Technical Writer and Editor/Licensing and Nuclear Safety/Repository Project Management
Bechtel SAIC Company, LLC
1180 North Town Center Drive
Las Vegas, Nevada 89134

Dr. Peter SWIFT
Office of the Chief Scientist
Lead Laboratory/Sandia National Laboratories,
MS 0778
Albuquerque, NM 87185-0778

EUROPEAN COMMISSION (EC)

Mr. Gérard BRUNO
DG TREN/H2
European Commission
Bat. EUFO 4289
10 Rue Robert Stumper
2557 Luxembourg
Luxembourg

Mr. Wolfgang HILDEN
DG Transport and Energy, Safety and Radioactive Waste Group
European Commission
Complexe Euroforum
10 rue Robert Stumper
2557 Luxembourg
Luxembourg

Dr. Michel RAYNAL
DG-RTD/J2
European Commission
CDMA 01/61
1049 Brussels
Belgium

M. Simon WEBSTER
DG-RTD, Nuclear Fission and Radiation Protection
European Commission
CDMA 01/70
1049 Brussels
Belgium

INTERNATIONAL ATOMIC ENERGY AGENCY (IAEA)

Mrs. Eliana AMARAL
Director, Division of Radiation Transport and Waste Safety
Division of Radiation, Transport
Wagramerstrasse 5
1400 Vienna
Austria

M. Didier LOUVAT
Nuclear Safety and Security
Agence internationale de l'énergie atomique (AIEA)
Wagramer Strasse 5
1400 Vienna
Austria

Mr. Phil METCALF
International Atomic Energy Agency
Wagramer Strasse 5
Vienna
Austria

OECD NUCLEAR ENERGY AGENCY (OECD/NEA)

Mr. Torsten ENG
OECD/NEA/DEV
2 rue André-Pascal
75016 Paris
France

Ms. Elizabeth FORINASH
Administrator, Radioactive Waste Management
OECD/NEA/PR
2 rue André-Pascal
75016 Paris
France

Ms. Gail MARCUS
OECD/NEA
2 rue André-Pascal
75016 Paris
France

Mr. Federico MOMPEAN
OECD/AEN/DB
2 rue André-Pascal
75016 Paris
France

Mr. Claudio PESCATORE
Principal Administrator for Radioactive Waste Management
OECD/NEA/PR
2 rue André-Pascal
75016 Paris
France

Mr. Hans RIOTTE
Head, Radiation Protection and Waste Management Division
OECD/NEA/PR
2 rue André-Pascal
75016 Paris
France

CONSULTANTS

Dr. Johan ANDERSSON
JA Streamflow AB
Vallvägen 22
12533 Älvsjö
Sweden

Dr. David BENNETT
TerraSalus Limited
3A Orchard House
Church Lane
LE15 9EL Bisbrooke
United Kingdom

Mr. Gyula DANKO
Golder Associates (Hungary) Ltd.
Huvosvolgyi ut 54.
H-1021 Budapest
Hungary

Dr. Daniel GALSON
Managing Director
Galson Sciences Limited
5 Grosvenor House
Melton Road
LE15 6AX Oakham, Rutland
United Kingdom

Mr. Martin GOLDSWORTHY
Golder Associates GmbH.
Vorbruch 3
29227 Celle
Germany

Mrs. Ann MACLACHLAN
European Bureau Chief
Platts Nuclear Publications
11/13, avenue de Friedland
75008 Paris
France

Dr. Charles MCCOMBIE
McCombie Consulting
Enzberghöhe 2
5073 Gipf-Oberfrick
Switzerland

Mr. Trevor SUMERLING
Safety Assessment Management. Ltd
Beech Tree House
Hardwick Road
RG8 7HW Whitchurch-on-Thames
United Kingdom

OECD PUBLICATIONS, 2 rue André-Pascal, 75775 PARIS CEDEX 16
Printed in France.