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The Safety Case for Deep Geological Disposal of Radioactive Waste: 2013 State of the Art

Symposium Proceedings
7-9 October 2013
Paris, France

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Radioactive Waste Management Committee

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ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

Foreword

In 2007, the Nuclear Energy Agency (NEA), in concert with the International Atomic Energy Agency (IAEA) and the European Commission (EC), organised a Symposium, entitled *"Safety Cases for the Deep Disposal of Radioactive Waste: Where Do We Stand?"* (NEA, 2008). Since then, there have been major developments in a number of national geological disposal programmes and significant experience has been obtained in preparing and reviewing cases for the operational and long-term safety of proposed and operating geological repositories. Especially, three national programmes are now, or will shortly be, at the stage of licence application for a deep geological repository for the disposal of spent nuclear fuel or high-level and other long-lived radioactive waste.

Thus, the purpose of this Symposium, *"The Safety Case for Deep Geological Disposal of Radioactive Waste: 2013 State of the Art"*, was to assess the practice, understanding and roles of the safety case, as applied internationally at all stages of repository development, including the interplay of technical, regulatory and societal issues, as they have developed since 2007.

The Symposium was organised and hosted by the Nuclear Energy Agency (NEA) of the Organisation for Economic Co-operation and Development (OECD), and co-sponsored by the EC and the IAEA.

The 2013 Symposium attracted 168 participants from 65 organisations and 17 countries and international bodies.

The Symposium was chaired by Klaus-Jürgen Röhlig, who is also the Chair of the NEA Integration Group for the Safety Case (IGSC). The planning and implementation of the symposium were supported by the efforts of staff of the NEA and of the programme committee.

This synopsis was drafted by Trevor Sumerling (Safety Assessment Management Ltd, UK), finalised under the direction of the Chair and Gloria Kwong of the NEA and approved by the programme committee.

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Synopsis

Background

The concept of a “safety case” for a deep geological repository for radioactive waste was introduced by the OECD/NEA Expert Group on Integrated Performance Assessment (IPAG) and further developed in an OECD/NEA report on *Confidence in the Long-Term Safety of Deep Geological Repositories* (1999). Since then the concept has been elaborated on in NEA documents describing the nature and purpose of safety cases (OECD/NEA, 2004, 2013a) and has been the basis of NEA/IGSC collaborative projects (OECD/NEA, 2009, 2011) and EU projects (EC, 2011). The concept has been taken up in international safety standards as promulgated by the IAEA (2006, 2011) and recently in ICRP recommendations on the application of the system of radiological protection to geological disposal (ICRP, 2013).

Meanwhile, the concept has been developed practically and applied within many national radioactive waste disposal programmes and also taken up in some national regulatory guides. The NEA has used the concept as a guide in several international peer reviews of national repository programmes and safety documentation. In Europe, the EU Directive 2011/70/Euratom (EU, 2011) establishes a framework to ensure responsible and safe management of spent fuel and radioactive waste by member states that, *inter alia*, requires a decision-making process based on safety evidence and arguments that mirrors the safety case concept.

Aims of the symposium

The purpose of this symposium, *2013 State of the Art*, was to assess the practice, understanding and roles of the safety case, as applied internationally at all stages of repository development, including the interplay of technical, regulatory and societal issues, as they have developed since 2007. In particular, the symposium aims were:

- to share experiences on preparing for, developing and documenting a safety case from both the implementer’s and reviewer’s perspectives;
- to share developments in requirements, expectations and experience gained in judging the adequacy of safety cases;
- to identify issues that may arise as repository programmes mature;
- to understand the importance of a safety case in promoting and gaining societal confidence;
- to gain experience from other fields of industry and technology in which concepts similar to the safety case are applied;
- to receive indications useful to the future working programme of the NEA and other international organisations.

Symposium programme

The symposium was organised into main plenary sessions covering:

- international activities and experience related to the safety case since 2007, including perspectives from the EC, IAEA and NEA;
- national safety case presentations at different stages of programme development from implementer and regulator perspectives.

These were followed by special topical plenary presentations on:

- staff management, training and knowledge management;
- issues and long-term governance of CO_2 storage.

This was followed by parallel sessions on:

- specific issues and challenges in safety case development;
- performance and safety assessment;
- science and technological basis;
- broader perspectives.

Allowance was also made for informal presentation and discussion of poster papers.

The final day of the symposium was devoted to a plenary session on the societal context of the safety case, followed by rapporteur reporting, open discussion and closing remarks from the Chair.

A session plan, list of the papers and abstracts are available from the NEA IGSC website. Key observations and findings from the symposium are summarised in the following sections.

Safety cases at different stages of repository development

As anticipated in the NEA safety case documents, the extent and level of detail of safety cases evolve as repository programmes develop.

Building a safety case in advance of concept and site decisions is challenging, but can be used to indicate preferred site qualities and as a basis for waste management decisions (e.g. concerning waste packaging) (*Bailey, Ute*). Working with data from a site (even a site that has been ruled out) provides an opportunity to practice the processes of data collection, interpretation and synthesis into performance assessment, as has been shown in the past at the Canadian URL, Äspo, Mol and many other locations, and at this symposium in the case of the Korean Underground Research Tunnel (KURT) (*Jeong*).

Several of the national safety case presentations illustrated the long-term commitment to research organisations and facilities, including generic and site-specific underground facilities (OECD/NEA, 2013b), that is needed to develop the scientific and technical understanding that is key to a science- and knowledge-based safety case.

As the safety case is developed, R&D programmes should be aligned with the specific understanding and data needed to support the safety case. For example, safety functions, and related safety requirements, statements, and arguments to support safety can provide a framework to prioritise R&D and site characterisation activities (*Capouet*). As programmes progress to the phase of demonstration of construction and emplacement of the engineered barrier systems, a wider range of technological tests is required, with resource and cost implications. Co-operative sharing of research facilities, skilled human resources and costs is both efficient and necessary, e.g. as fostered in the EC Implementation of Geological Disposal Technology Platform (IGD-TP) (*Andra, SKB, POSIVA, van Geet*).

Presentations on the role of the safety cases in support of a site selection process (*Zuidema, Leuz*) identified the importance of clear process and responsibilities, and involving all stakeholders, as key ingredients to success. Decisions need to rest with agreed actors. The concurrent development of a safety case and regulations presents a specific challenge, in particular balancing the desire to establish binding rules while also allowing for flexibility. Safety cases supporting the siting process must address both technical requirements and societal expectations. The implementer has the role of gathering technical evidence to demonstrate safety, the regulator has the role of verifying that the proposed disposal system complies with all applicable regulatory requirements, and potentially affected communities need to have opportunities to express their views and receive the information they need to satisfy their concerns. Iterative development of safety cases provides both feedback to repository design (*Wollrath*) as well as increased mutual understanding between implementer, regulator and host communities.

To be accepted as a location for development of a geological repository, a site must pass technical and societal tests. Societal aspects and ingredients of a site selection process are discussed below. Key, however, is that the safety case must be sufficiently broad to meet the information needed by all actors in the decision-making process and wider audiences that may be involved (*Boissier, Hocke*). This will include quantitative demonstrations of performance and safety, and qualitative arguments in support of safety and site and design preferences (*Soderblom*). In both cases, uncertainties must be recognised and critically examined in the safety case (*Tweed*). The balance between qualitative arguments and simple calculations shifts towards more rigorous quantitative and site- and design-specific arguments and evidence as programmes progress (*Zuidema*). Some crucial safety components may require considerable time to be researched, developed and qualified, and it is important that allowance be made for this.

In Sweden and in Finland, and soon in France, the responsible waste management organisations have made, or will shortly make, applications for a licence to construct a deep geological repository for the disposal of spent fuel, or HLW and other long-lived radioactive waste (*Hedin, Dverstorp, Vira, Heinonen, Boissier, Tanguy*). The legal and decision-making frameworks in which they work are different, however; this means that each case may differ in terms of what has been applied for or granted (for example regarding the level of detail in the technical design or future decision points).

As projects move towards licensing and practical realisations, aspects of constructability, mining safety and operational safety must also be addressed (*Boissier, Hedin, Vira*). Mining and operational safety may be considered with long-term safety in a single safety case submission or be the subject of separate, parallel submissions, depending on the national legal and regulatory requirements. In any case, development of a design that fulfils all safety requirements, and that can be realised at the industrial scale, is a challenge in which the long-term safety case is only one input. The IGSC has started to take account of these trends, e.g. in the Expert Group on Operational Safety initiated in 2013 (*Röhlig IGSC presentation*).

For the above reasons, there is not a single unified format for all safety cases. It is important, however, that the internationally accepted safety case components can be mapped to the content of national documents, e.g. of licence applications.

In some countries the timetable for applications and decisions related to the development of a geological repository is mandated by government; in other countries, while a process may be defined, it falls on the implementer to bring forward proposals at what they judge to be suitable maturity. In Europe, the EU Directive 2011/70/Euratom requires that governments set out their timetable for implementation of safe management of spent fuel and radioactive waste by 2015 (*Blohm-Hieber*). In any case, the regulators play a key role in providing clear regulations and guidance to the implementer on expectations at a given programme step, plus reviews of safety case quality and readiness for moving forward (*Dverstorp*). While the regulator's statutory role is to advise government, the exercise of an open and critical technical review by the regulator is key to protecting the interests of the local community and providing reassurance to the wider public.

In undertaking deep underground investigations and excavations, but also in continuing research, unexpected discoveries are to be expected. The repository design and safety case must be sufficiently flexible to respond and adapt to results of investigations and monitoring, including unexpected findings. The uncertainty and potential for unexpected findings must be acknowledged and the flexibility of the disposal concept communicated beforehand. Thus, all parties (implementer, regulator, local communities and other stakeholders) should understand the scope for project modification and that not every unfavourable finding leads to abandoning a site.

Experience from the WIPP facility (*Patterson, Peake*) provides an example of the periodic re-licensing that is current practice at all operating nuclear facilities, as applied to a geological repository. The legally-bound exchange between the implementer and regulator(s) assures that the facility continues to meet all safety requirements, while also giving the possibility for optimisation of disposal operations drawing on the practical experience at the facility.

Knowledge management

The importance of collaboration between different scientific and technical teams engaged in repository programmes and associated R&D, and the retention of knowledge and expertise over the several decades or more during which a repository is developed and operated, were mentioned as key concerns in several programmes. The issues were addressed specifically in a presentation that described the IGSC-initiated project on "Staff Management, Training and Knowledge Management" (*Makino*).

Staff management, training and knowledge management are organisational issues that are critical in any large, collaborative scientific and technical projects. Just as it must be ensured that the financial resources are available to carry through a geological repository project, so it must be ensured that the knowledge and expert personnel are (and continue to be) available and further developed over the period of the project. For geological repositories, this is a particular challenge since the projects typically are and will be carried out over more than a century. Thus, these issues are directly relevant to the safety case and confidence in the safety case.

The issues of knowledge management and maintenance of expertise may be especially acute for regulatory bodies, since they have fewer staff than implementers, and for their reviews need to call on scientific and technical experts who can be regarded as independent, i.e. have not been involved in the development of the safety case under review. Some regulators overcome this by maintaining a strong parallel programme of research and assessment (*Dverstorp, Tanguy*). Regulators may also rely on participation in international projects and exchanges (*Serres*), and engaging independent experts as needed (*Dverstorp*).

The importance of staff management, training and knowledge management is recognised in IAEA publications on management systems (IAEA, 2006b, 2008) and also in several national regulations. The RWMC Forum on Stakeholder Confidence (FSC) has identified features such as highly motivated staff, learning capacity, high levels of skill and competence, specific management plans and transparency of processes, responsibilities and behaviours, as key ingredients towards confidence building in a safety case.

These issues will be further worked on within a continuation of the IGSC initiative on "Staff Management, Training and Knowledge Management" (*Makino*), and within the IGD-TP (*van Geet*).

It can also be valuable to exchange knowledge and experience with other technical areas that share common aspects with geological disposal. For example the keynote presentation on long-term governance of carbon dioxide storage (*Farret*) explored issues also of concern in geological disposal, including the requirement for long-term containment, dealing with geological uncertainties, short and long-term monitoring, long-term governance and societal acceptance.

Societal context of the safety case

The meaning of safety and also acceptability of a development at a given location is a matter of societal view (*Pescatore*). Thus, although development of a safety case for a geological repository is a scientific and technical activity, it is also input for a societal decision. In developing and presenting a safety case the process of development and its perceived rigour and honesty are crucial to societal confidence and hence support for the outcomes.

The scientific and technical evidence that underpins the safety case is complex, but this is not a bar to societal participation. It is important that the evidence is fully and fairly presented, including negative aspects and uncertainties, so that the balance of evidence can be understood and assessed.

Trust is needed in processes, institutions and individuals. Trust takes time to win, for example through openness, clarity and willingness to respond to questions over extended dialogue, and can be easily lost.

A clear process must be defined by which choices and decisions will be made. The process should be open and transparent, with clarity about how the dialogue will inform the decisions to be made (*Soderblom*). Stakeholders who wish to be involved should be encouraged to do so, and be able to see that account is taken of their views.

Participants should have time to think issues through, and to become informed through reliable and balanced resources. This may include the provision of "protected spaces" (*Tweed*) or fora for reflective and informed discussion between stakeholders with a range of views and values. Structured dialogue between policy-makers, regulators, technical experts and non-technical participants can enable all to re-evaluate their perspectives and assumptions in the light of those of others, evolve their thinking and develop mutual and convergent understanding.

The active participation of a host community or municipality for a repository provides multiple benefits. As well as providing practical advice and stating preferences regarding details of above-ground works and activities, the community provides a non-technical oversight of the project over its lifetime, which is a social validation for the project. To do this, the community should take an active part in the decision-making process, from start to closure (*Soderblom*).

The conditions for regulatory permission, and the degree of commitment and remaining capacity for withdrawal must be stated clearly at each decision point, so that the community's representatives can make informed decisions (*Leuz*). The process should be guaranteed by government, and possibly in legislation, so there is no "change of rules" that can undermine previously given commitments.

State of the art 2013

The 2013 Symposium showed that a clear understanding of the technical components of a safety case, as documented in the recent safety case brochure, already exists (OECD/NEA, 2013).

Since 2007, further technical experience has been gained from the iterative and stepwise processes, e.g. on aspects of incorporating new data; model development; safety concept and design refinement; and performance assessment and documentation. Several national programmes now have moved into licensing stages, which has opened up new topics to be discussed.

Examples given in the symposium illustrate how both the safety case content and its presentation have been enhanced based on technical experience, audience responses, and dialogue. These include regulator-implementer interactions, engagement with local communities and peer reviews of safety cases and repository programmes.

There are differences in the realisation and roles of a safety case, in different countries and at different times (e.g. some safety cases focus on post-closure whereas others programmes may also cover operational safety), depending on the stage of the programme and the decision they inform and taking into account the different legal, regulatory and social requirements. For these reasons, there is not a single unified format for all safety cases. It is important, however, that the internationally accepted safety case components can be mapped to the content of national documents, e.g. licence applications. Nevertheless, there is an international consensus on elements and methodology especially for assessing long-term safety as described in IAEA Safety Standards and in OECD/NEA documents (2011), which is visible in most national documents.

At its core, the development of a safety case for a geological repository is a scientific and technical activity, based on rigorous tools and processes. The safety case, however, must also satisfy regulatory and societal requirements. Thus, the safety case can be seen as a "scientific and technical activity embedded in a societal context", the latter being particularly important for confidence building.

In practice, proposals for a geological repository and the accompanying safety case are developed iteratively. Mutual understanding and confidence in the overall outcome and processes will be improved through ongoing dialogue between the implementer, regulator and other stakeholders, especially representatives of potential host communities or municipalities. Through this dialogue, regulators and other stakeholders can influence the decisions and final outcomes, as well as ensuring the safety case contains the information for their needs. The concerns of each stakeholder need to be taken into account so that joint understanding is established and expectations converge.

Over the several decades in which the repository programme evolves, the safety case will go through a number of iterations. Newly gained knowledge, including remaining uncertainties, must be clearly communicated with all stakeholders. Each time the safety case is revised, new needs (R&D, technological development, demonstrations) as well as areas of improvement of the disposal programme will be identified. These improvements will allow the design of the disposal system to be optimised. Through these iterations, robustness of the disposal solution has to be improved, unexpected findings have to be addressed, and the safety case has to be strengthened, leading to increased confidence in the safety of the disposal solution.

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Session 3

National Safety Case Presentations

Compiling and reviewing the safety case at different stages of repository development

Session 3.1

Building Generic Safety Cases: Examples from the United Kingdom, the United States and Korea

Developing a generic environmental safety case

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The Nuclear Decommissioning Authority (NDA) has been charged with implementing the United Kingdom government's policy for the long-term management of higher activity radioactive waste by planning, building and operating a geological disposal facility (GDF). Within the NDA, we – the Radioactive Waste Management Directorate (RWMD) – are tasked with the development of a GDF. The UK government has also decided that a process of voluntarism and partnership will be followed to identify a suitable site for the GDF. To date there is no volunteer community and the site selection process to find a volunteer host community is under review.

RWMD has an ongoing role to provide advice to UK radioactive waste producers on the conditioning and packaging of wastes and to undertake disposability assessments of waste packaging proposals to determine their suitability for eventual disposal in a GDF. We also need to demonstrate our confidence that a GDF would be safe. Therefore RWMD has published a generic Environmental Safety Case (ESC) (NDA, 2010) to demonstrate that we are confident that a GDF could be developed to meet the guidelines set down by the environmental regulators (EA/NIEA, 2009) in a range of geological settings. The ESC includes reference case calculations that are used as a benchmark for disposability assessments.

The challenges of developing a generic ESC

The UK has a range of different geological settings that could potentially be suitable for siting a GDF. In terms of potential host rock, these include a higher strength rock (for example, granite), a lower strength sedimentary rock (for example, clay) and evaporites (salt). The optimum GDF design for each of these geological settings would be different, therefore, as the geological setting is unknown, the GDF design and the types of engineered barriers employed to provide safety are also unknown. Uncertainty over both the geological setting and the GDF design present considerable challenges for developing a generic ESC.

Furthermore, the UK has a wide range of higher activity nuclear materials that may need to be disposed in a GDF. These materials are listed in Table 1. Not all these materials are currently declared as wastes (for example spent nuclear fuel, separated plutonium and uranium are regarded as "zero value assets" in the UK). However, to be fully embracing of potential outcomes, the generic ESC considered the full inventory listed in Table 1.

The approach adopted for the generic ESC was to identify a set of concept examples for each of the three generic geological settings identified. For each geological setting, two illustrative example disposal concepts were identified: one for intermediate-level waste (ILW), low-level waste (LLW) and depleted, natural and low-enriched uranium (DNLEU); and one for high-level waste (HLW), spent nuclear fuel and plutonium. These illustrative disposal concept examples were based on those that are being developed or implemented around the world in the potential host rocks, and are listed in Table 2.

Table 1: Inventory considered in the generic ESC

Materials	Packaged volume		Radioactivity (at 1 April 2040)	
	Cubic meters	%	Terabequerels	%
HLW	1 400	0.3%	36 000 000	41.3%
ILW	364 000	76.3%	2 200 000	2.5%
LLW (not for LLWR)	17 000	3.6%	<100	0.0%
Spent nuclear fuel	11 200	2.3%	45 000 000	51.6%
Plutonium	3 300	0.7%	4 000 000	4.6%
Uranium	80 000	16.8%	3 000	0.0%
Total	476 900	100%	87 200 000	100%

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Table 2: Illustrative geological disposal concept examples used in the generic ESC

Host rock	Illustrative geological disposal concept examples[d]	
	ILW/LLW	HLW/SF
Higher strength rocks[a]	UK ILW/LLW Concept *(NDA, UK)*	KBS-3V Concept *(SKB, Sweden)*
Lower strength sedimentary rock[b]	Opalinus Clay Concept *(Nagra, Switzerland)*	Opalinus Clay Concept *(Nagra, Switzerland)*
Evaporites[c]	WIPP Bedded Salt Concept *(US-DOE, USA)*	Gorleben Salt Dome Concept *(DBE Technology, Germany)*

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[a] Higher strength rocks – the UK ILW/LLW concept and KBS-3V concept for spent fuel were selected due to availability of information on these concepts for the UK context.

[b] Lower strength sedimentary rocks – the Opalinus Clay concept for disposal of long-lived ILW, HLW and spent fuel was selected because a recent OECD Nuclear Energy Agency review regarded the Nagra (Switzerland) assessment of the concept as state of the art with respect to the level of knowledge available. However, it should be noted that there is similarly extensive information available for a concept that has been developed for implementation in Callovo-Oxfordian Clay by Andra (France), and which has also been accorded strong endorsement from international peer review. Although we will use the Opalinus Clay concept as the basis of the illustrative example, we will also draw on information from the Andra programme. In addition, we will draw on information from the Belgian supercontainer concept, based on disposal of HLW and spent fuel in Boom Clay.

[c] Evaporites – the concept for the disposal of transuranic wastes (TRU) (long-lived ILW) in a bedded salt host rock at the Waste Isolation Pilot Plant (WIPP) in New Mexico was selected because of the wealth of information available from this United States Environmental Protection Agency (EPA) certified (and operating) facility. The concept for disposal of HLW and spent fuel in a salt dome host rock developed by DBE Technology (Germany) was selected due to the level of concept information available.

[d] For planning purposes the illustrative concept for depleted, natural and low-enriched uranium is assumed to be the same as for ILW/LLW and for plutonium and highly-enriched uranium is assumed to the same as for HLW/SF.

Each of the six example disposal concepts listed in Table 2 is described in the generic ESC in terms of the multiple barriers and safety functions it uses to isolate and contain the waste materials.

Perhaps the greatest challenge in developing a generic safety case, especially one that is required to provide a benchmark reference numerical evaluation of safety for disposability assessments, is considering how to define such an evaluation that is suitably generic but still has sufficient meaning to provide useful information as a benchmark for a wide range of disposability assessments. Other significant challenges included communicating the meaning of the generic safety case and particularly its role in disposability assessments.

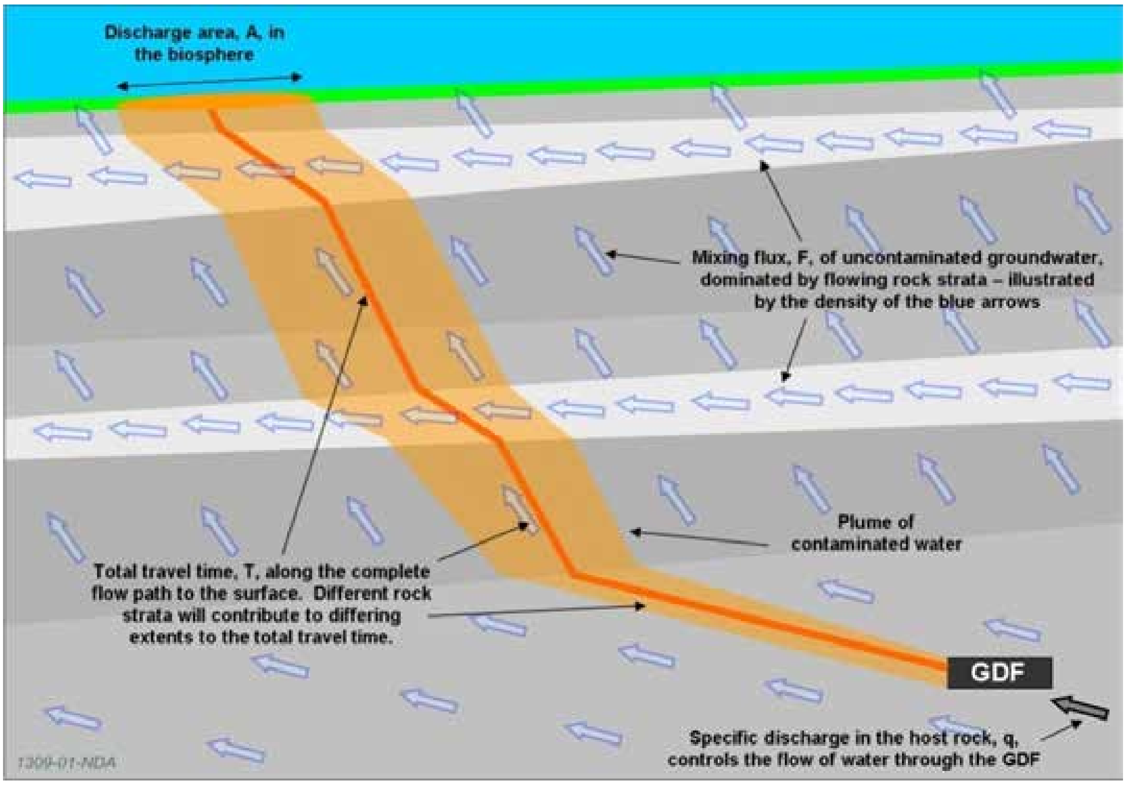
Various approaches were considered for dealing with an unknown geological setting, including using real data to present example calculations for a range of actual sites: however, the disposability assessment process wanted a single reference benchmark in which the calculated risks are likely to be conservative – that is, for an actual site, the calculated risks would be unlikely to be greater than those calculated for the reference models and reference parameter values. This approach provides confidence that if wastes are conditioned and packaged in such a way as to satisfy the regulatory risk guidance level (EA/NIEA, 2009) on the basis of the reference model, they should be suitable for disposal at any likely actual GDF site.

For a GDF situated in water-saturated rocks where there is groundwater movement, the dissolution and transport of radionuclides in groundwater is likely to be the most significant mechanism by which radionuclides could eventually return to the surface environment and give rise to risks to humans. The groundwater flow field at a particular site will be highly dependent on the rock strata, the hydrogeological properties of those rock strata and the natural driving forces for groundwater movement. The three illustrative generic geological settings considered to be relevant to the UK will have very different groundwater flow field properties. In evaporites there will be little or no mobile groundwater present in the host rock and hence radionuclide transport by groundwater would not be a significant issue. Lower strength sedimentary rocks in the UK are likely to be water-saturated, but may be of sufficiently low permeability that there is little groundwater movement and radionuclide transport is likely to be dominated by generally slower diffusion processes. Higher strength rocks are generally water-saturated and may contain relatively permeable rock strata and/or fractures through which significant groundwater movement can occur.

Therefore, in order to be conservative in our approach, we chose to base our model representation on a stylised flow field for a higher strength host rock setting. We defined four parameters, as illustrated in Figure 1, to represent the properties of a generic groundwater flow field. When combined with the transport and retardation properties of specific radionuclides, these parameters determine the performance of the geological barrier in our generic assessment of radionuclide transport by groundwater.

We also need to represent the performance of the engineered barrier system (EBS), again in the absence of any site-specific EBS design. In terms of the impact on radionuclide transport in groundwater, an important property of the EBS is the period over which radionuclides are contained by the engineered barriers. On this basis, a fifth parameter, C, was defined as the time for which the waste container provides complete containment of radionuclides. Again, a conservative approach was taken to assigning reference parameter values for C. Since most ILW containers are vented to allow the escape of gas, for these wastes C was assigned the value zero. In contrast, we expect the container materials under consideration for HLW and spent nuclear fuel to provide a significant period of containment (up to hundreds of thousands of years) and this is reflected in the parameter ranges for C for such wastes.

Figure 1: Schematic of groundwater pathway conceptualisation in generic ESC



Groundwater parameter definitions:

q – The specific discharge through the undisturbed host rock at the GDF location. This provides a measure of how the rate of groundwater flow through the engineered barriers could affect calculated results.

T – The groundwater travel time from the GDF to the surface environment. This provides a measure of how the containment capability of the geological environment could affect the calculated results.

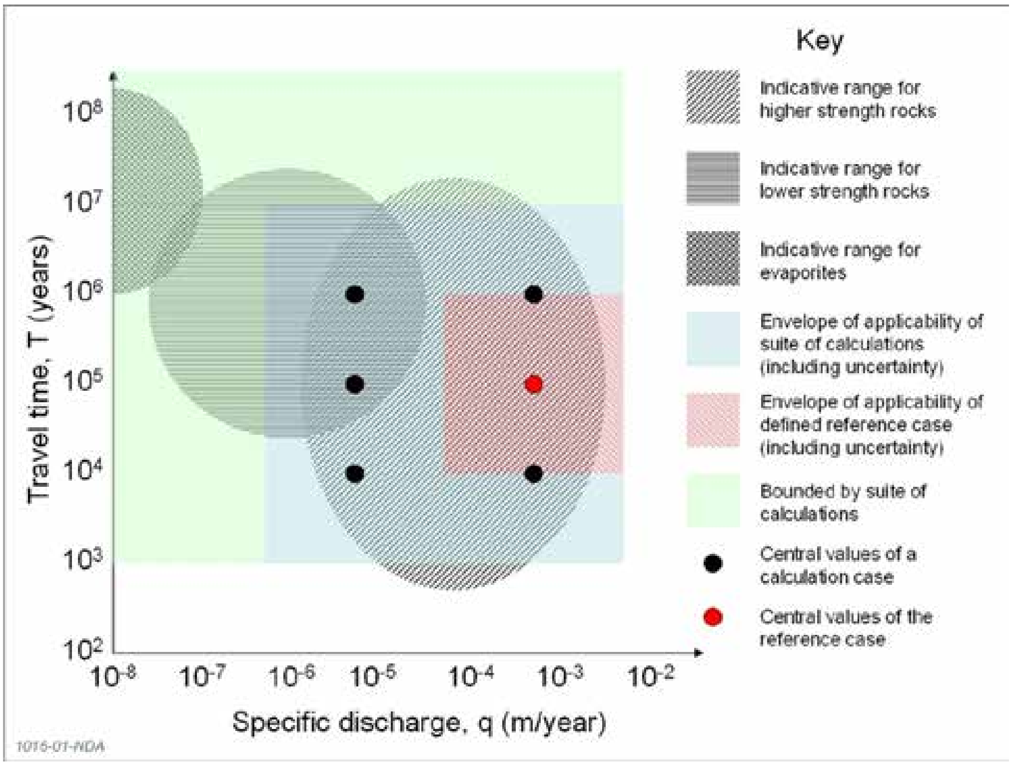
F – The groundwater mixing flux in the overlying rocks. This provides a measure of how the dilution potential of geological units overlying the host rock could affect the calculated results.

A – The discharge area into which the contaminant plume is released at the surface. This provides a measure of how the geosphere-biosphere interface could affect the calculated results.

The values for q, T, F, A and C were sampled in a probabilistic calculation and in addition a number of sensitivity calculations were performed. Figure 2 indicates the range of values considered for the parameters q and T in the generic ESC and how these relate to the generic host rock settings. These indicative ranges were derived from a study of groundwater modelling approaches for different generic geological environments (Towler, 2008).

The shaded blue area in Figure 2 indicates the envelope of (q, T) parameter space covered by the suite of calculations performed in the generic ESC. It can be seen that this encompasses virtually all the (q, T) parameter space identified as likely for higher strength rocks in the UK (indicated by the diagonally shaded ellipse in Figure 2). Furthermore, this envelope of calculations also bounds (as indicated by the green shaded area) the indicative (q, T) parameter ranges for lower strength rocks and evaporites. On this basis, the groundwater pathway calculations in the generic ESC are judged to be conservative and bounding for the likely range of GDF host rock settings in the UK.

Figure 2: Relationship of values of q and T to different host rocks



Proposed future developments

Peer review comments on the generic ESC indicated that for some people the rationale for the generic assessment approach was difficult to comprehend. There is a clear limit on the extent to which meaningful calculations can be carried out at a generic stage, when the site and concept are unknown. The generic ESC calculations for a wide range of parameter combinations gave rise to risks both above and below the regulatory risk guidance level. However, the generic ESC stresses that it is the *relative* contributions to risk from different components of a GDF and from different radionuclides in the inventory that are of most interest, rather than the *precise* calculated risk values. Once we have information about a site, we will develop new, appropriate models for the concept that we plan to implement, based on a site descriptive model of the geological setting and the groundwater flow field.

However, until we have confidence in a particular site, it is our strategy to continue to develop and maintain our generic ESC, in parallel with any future developing site-specific ESC (NDA, 2012), to provide continuity and robustness to our disposability assessment process. In planning for our next update to the generic ESC, in response to peer review feedback, evolving research and the changing needs of the UK programme, we are considering ways of improving the presentation of our ESC. This may include less emphasis on the numerical calculations, whilst still providing a reference benchmark for disposability assessments.

We are developing a “safety narrative” that explains how safety is provided in terms of the different safety functions provided by the various natural and engineered barriers. This safety narrative will also explain how the safety functions operate over different time frames and we are exploring ways of presenting the safety case that highlight the performance of the safety functions and also the uncertainties associated with that performance over different time frames. The aim is to build confidence in our understanding of the system and to aid communication with a variety of technical and non-technical audiences.

It is proposed that we will still consider generic concepts, based on those developed in the UK and internationally, for different geological settings. Our safety narrative will explain how the safety functions of the features designed into each concept work with the natural safety functions of the geological environment to isolate and contain radionuclides over different time frames. The majority of radionuclides will be sufficiently contained that they decay within the engineered barriers. However, for some long-lived radionuclides total containment within the engineered barriers cannot be guaranteed for all time frames. For these radionuclides, the retardation properties of both the engineered and natural geological barriers act to delay and disperse the transport of radionuclides to ensure that should they reach the surface environment, they will only do so in concentrations that are consistent with satisfying the regulatory risk guidance level (EA/NIEA, 2009).

It is proposed that the generic ESC will include a range of safety arguments, supported by appropriate calculations (using deterministic, probabilistic and insight models) to demonstrate how safety is achieved over different time frames. For appropriate time frames it is proposed that this will continue to include reference calculations for the groundwater pathway and other relevant release pathways. However, proposals are being developed that consider when it may be appropriate to cease to present means of probabilistic calculations on the basis that such calculations are only meaningful in time frames where the relevant uncertainties can be reasonably quantified. For later time frames (for example, beyond a hundred thousand years or so), it may be more appropriate to use more qualitative arguments or consider stylised “What if?” calculations to represent different potential evolutions of the GDF system over very long time scales.

It is our intention to discuss these proposals with our regulators and a range of stakeholders, prior to developing and publishing our next generic ESC.

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Using safety assessment techniques to build confidence in repository performance: The United States experience

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Introduction

The United States Department of Energy (US DOE) has prepared detailed safety assessments (called performance assessments or total system performance assessments in the US regulations and practice) during the last 20 years for two deep geologic repositories for the disposal of radioactive wastes: the Waste Isolation Pilot Plant (WIPP) in south-eastern New Mexico, and the proposed Yucca Mountain repository in Nevada for spent nuclear fuel and high-level radioactive waste. The DOE is currently conducting generic safety assessment analyses of disposal concepts that are potentially viable alternatives for the US in the future, and anticipates that performance assessments will continue to be an important part of the process of evaluating the suitability of disposal alternatives.

For both the WIPP and the proposed Yucca Mountain repository, the performance assessment was an essential part of the DOE's application to the regulator, which in the US serves the same function as the safety case as defined by the international radioactive waste disposal community (OECD/NEA, 2004). For both sites, performance assessments provided quantitative estimates of the long-term performance of the disposal system (10 000 years for WIPP and 1 000 000 years for Yucca Mountain), and at both sites the performance assessments matured through a series of iterations over a decade or more, with increasing sophistication in the data, models and methods used in the analyses. Current performance assessment analyses for hypothetical generalised disposal systems build on the experience gained from both the WIPP and Yucca Mountain programmes, and, although they are unavoidably simplified compared to those done for specific sites, they use the same basic approaches.

Confidence in the overall long-term safety of a disposal system comes through a sound understanding of the geologic and engineered barriers relied on to isolate the waste. Performance assessments can contribute to this confidence in multiple ways; for example, through clear demonstrations that the limits of scientific understanding are acknowledged and that unavoidable and irreducible uncertainties have been taken into account, through identification of the ways in which the individual components of the system work together to ensure robust isolation, and through quantification of the ways in which uncertainties impact estimates of overall performance. This paper focuses on three specific attributes of the US DOE's performance assessments that have contributed to overall confidence in long-term repository safety. First, the paper addresses an approach to ensuring a comprehensive treatment of uncertainty associated with the potential occurrence of rare but disruptive events that could impact repository performance in the future. Second, the paper addresses the importance of demonstrating how individual components of the repository and its surrounding geology work together to provide an effective system of engineered and natural barriers. Third, the paper reviews the benefits of a rigorous uncertainty analysis that allows quantification of the relative importance of the impacts of uncertainties in input parameters used in Monte Carlo simulations on the overall estimates of performance. These three attributes of a performance assessment, done well, provide a strong technical basis for answering three questions fundamental to any evaluation of overall safety. Has the analysis appropriately considered the range of scenarios for the future evolution of the repository? Does the analysis demonstrate how the components of the system will work together as barriers to isolate the waste under these scenarios? How does our uncertainty affect our understanding of the repository's future performance?

Scenario uncertainty

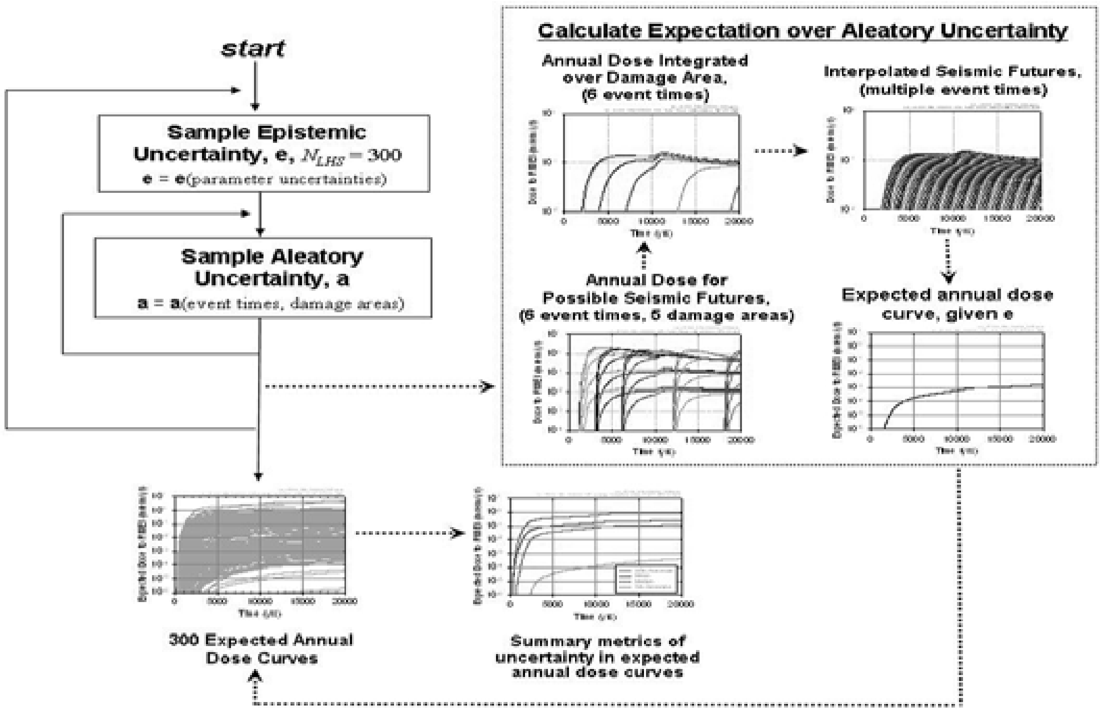
One of the important lessons learned in the US performance assessments is that the overall clarity and rigor of the analysis benefits from a formal distinction between aleatory uncertainties, which in general derive from uncertainty about the occurrence of future events, and epistemic uncertainties, which derive from incomplete knowledge about the physical properties of the system. Simplistically, aleatory uncertainties can be thought of as irreducible, in the sense that no amount of additional research will provide a definitive answer about whether an earthquake will occur on a given date in the far future. Epistemic uncertainties, on the other hand, are in principle reducible through further research, although in practice it is neither necessary nor possible to fully characterise all aspects of a complex system. Performance assessments must be able to display the impact of both aleatory and epistemic uncertainty on the understanding of overall system behaviour. Given the complexity of assessments of this scale, distinctions between aleatory and epistemic uncertainties are not absolute, and analysts should acknowledge that the categorisation of specific uncertainties as either aleatory or epistemic is at least in part a subjective decision made to provide specific insights into the performance of the system.

Historically, performance assessments have relied on two basic approaches to incorporating uncertainty, often without fully distinguishing between aleatory and epistemic factors. In one approach, uncertainty can be incorporated by analysing multiple scenarios representing different possible future states of the system. For example, analysts may choose to perform separate assessments of anticipated, or "most likely", performance of the system and of unanticipated or "degraded" performance. In the US, the most complete implementation of this approach has been in a series of performance assessments for the proposed Yucca Mountain repository conducted by the Electric Power Research Institute (2009), in which all uncertainty was incorporated through the use of multiple deterministic (i.e. single values for each model parameter) scenarios using a logic tree approach to investigate the impacts of alternative values for key parameters. Alternatively, analysts can design performance assessments in which all uncertainty is incorporated into a single scenario through multiple Monte Carlo realisations that each use a different sampling of values from distributions assigned to uncertain parameters in the model. Examples close to this endpoint of the spectrum include early performance assessments performed for the proposed Yucca Mountain repository (US DOE, 1998). As these examples show, there is no uniquely correct way to subdivide possible future states of the system into scenarios: instead, scenarios are chosen because they have explanatory power, and allow the analyst to focus efficiently on events and processes of interest.

Beginning in the early 1990s with performance assessments for the WIPP, the formal distinction between aleatory and epistemic uncertainty allowed analysts to focus the Monte Carlo uncertainty analysis on epistemic factors (e.g. incomplete knowledge of the material properties of the disposal system), while providing a clear display of the importance to estimates of future performance of the irreducible aleatory uncertainty (for the WIPP, primarily the uncertainties that derive from the regulatory-based assumption that future human drilling events will be random in time and space) (Helton and Marrietta, 2000). Adaptation of this approach into the Total System Performance Assessment for the proposed Yucca Mountain repository in the middle 2000s (SNL, 2008) provided the analysis with a rigorous approach to defining separate scenarios distinguished by the occurrence or non-occurrence of aleatory events (e.g. seismic or igneous disruption) while allowing the propagation of epistemic uncertainty into each scenario. Impacts of uncertainty could be displayed separately for each scenario and combined with mathematical rigor into the composite display of uncertainty about the estimate of total mean annual dose required by regulations.

Figure 1 illustrates this process for the calculation of uncertainty associated with the consequences of seismic ground motion. The analysis begins with a single sample of values drawn from the 387 epistemic uncertain parameters included in the analysis (not all epistemic parameters were used in all scenarios). This sampling is then paired with values for two aleatory uncertainties, the time of the event and its magnitude, represented here as an uncertain area of damage on the waste package surface. (The uncertain damage area was itself derived from modelling of kinematic effects of a range of ground motions on the waste package.) The two aleatory uncertainties were treated as discrete variables, with system performance estimated for six event times and five damage areas, conditional on the selected sample of values for the epistemically uncertain parameters. As shown in Figure 1, averaging over the damage areas and interpolating between event times allowed construction of a single "expected annual dose" for the selected sample of epistemic uncertainty. Latin hypercube sampling of the epistemic uncertainty and Monte Carlo replication of the analysis resulted in a set of 300 expected annual dose curves, each conditional on the underlying sample values. Display of the full set of 300 expected annual dose curves provided the required total mean annual dose with the associated epistemic uncertainty (US DOE 2008, Section 2.4.1.3).

Figure 1: Computational strategy for computing the expected annual dose and associated summary metrics for a seismic ground motion damage scenario



Source: US DOE, 2008, Figure 2.4-8.

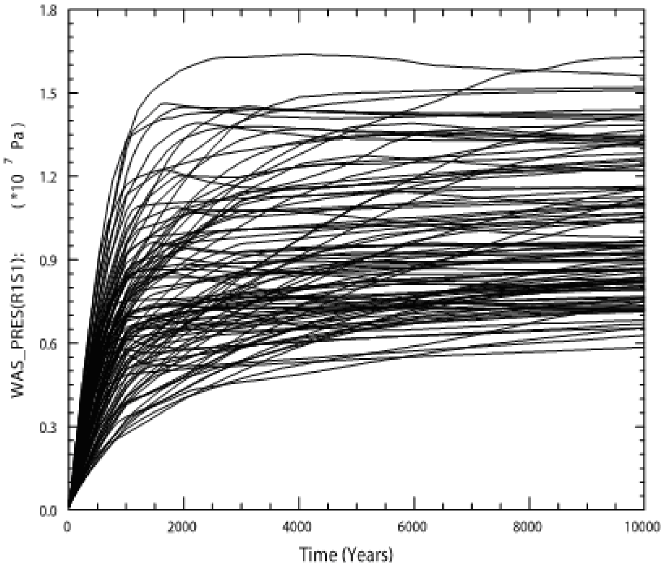
Understanding component and system barrier performance

A second important lesson learned from the DOE's iterative performance assessments is that the analysis must be capable of providing clear explanations of how the various components of the disposal system work together as barriers to ensure effective isolation of the waste. Barrier performance is typically characterised at the component level, in terms such as waste package lifetime, waste form degradation rate, or radionuclide transport times along groundwater flow paths. Confidence in the overall disposal system, however, requires understanding how the various components work together. A good performance assessment should display the behaviour of each component under the full range of conditions encountered in the analysis. In some cases, performance of a component within the system may be sufficiently independent from the rest of the disposal system that analyses can be meaningfully interpreted in stand-alone mode. For example, the potential effectiveness of retardation during radionuclide transport in flowing groundwater can be reasonably represented through the use of breakthrough curves calculated for hypothetical releases of unit masses. In other cases, however, the potential effectiveness of a component of the system may depend on the evolving characteristics of the surrounding environment, and it may not be simple to determine *a priori* what constitutes "better" or "worse" performance of the system. For example, elevated temperatures generally accelerate corrosion processes acting on metal waste packages, and therefore high temperatures in the near field environment are commonly viewed as detrimental to waste isolation. However, elevated temperatures may also delay resaturation of the near field environment, limiting the potential for water to reach the waste package at early times. Consideration of the performance of the near field environment and waste package in a coupled manner provides confidence that the barrier system will function under a range of conditions.

Figures 2 and 3 provide examples from the WIPP performance assessment of the importance of understanding barrier performance in the context of the full disposal system. In general, the exceptionally low permeability of the evaporite host rocks (halite and anhydrite) at WIPP is one of the most important attributes of the natural barrier system contributing to waste isolation. However, uncertainty in the true permeability of the host rocks (including the zone of disturbed halite that isolates the waste from relatively more permeable anhydrite layers above and below) coupled with uncertainty in the processes of brine flow and gas generation due to anoxic corrosion of iron and degradation of organic materials led to simulation of a broad range of modelled conditions. As shown in Figure 2, some samplings of epistemic uncertainty resulted in pressures within the undisturbed disposal region of 15 MPa and above, essentially equivalent to the lithostatic load at the repository depth. Other realisations showed peak pressures remaining below 6 MPa for the entire 10 000-year simulation. As shown in Figure 3, estimated saturation of the liquid phase (i.e. brine) in the waste disposal region showed a similarly complex range of outcomes. Most realisations showed a rapid rise in brine saturation as some liquid migrated into the disposal region, initially flowing toward the low pressure created by the excavation. In some realisations, brine saturation approached 1.0, but in most it remained below 0.5 at its peak and then fell as brine was consumed in the model by anoxic corrosion reactions. Gas generation, combined with the effects of creep closure of the excavated region, caused the pressure increase shown in Figure 2, reducing or even reversing the pressure gradient driving brine inflow, and brine saturations in Figure 3 stabilise in most realisations at values less than 0.2 (US DOE, 2009, Section PA-7.1).

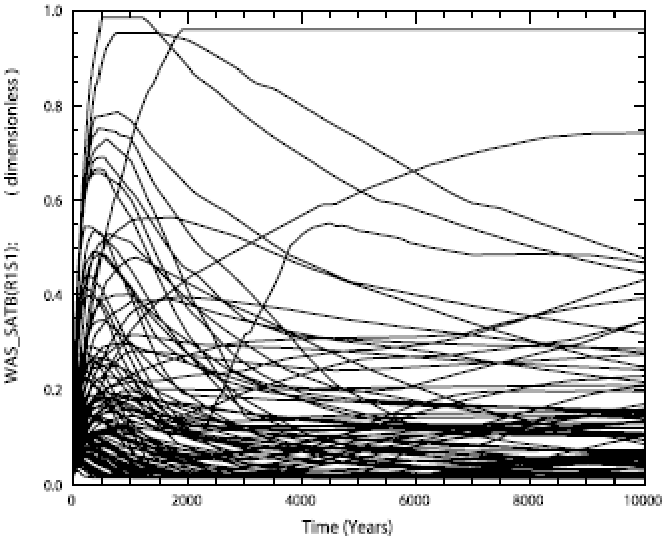
A full understanding of the contributions of the low permeability host rock at WIPP requires understanding the implications of the couplings that result in the complexity of Figures 2 and 3. In general, low permeability is a strongly positive attribute, limiting both the amount of water that can reach the waste and the potential for radionuclides to be transported away from the repository in water. However, high pressures can also contribute to increased releases in human intrusion scenarios, providing a mechanism to

Figure 2: Pressure in the waste disposal region at WIPP, 100 realisations of undisturbed performance



Source: US DOE, 2009, Figure PA-43.

Figure 3: Brine saturation in the waste disposal region at WIPP, 100 realisations of undisturbed performance



Source: US DOE, 2009, Figure PA-45.

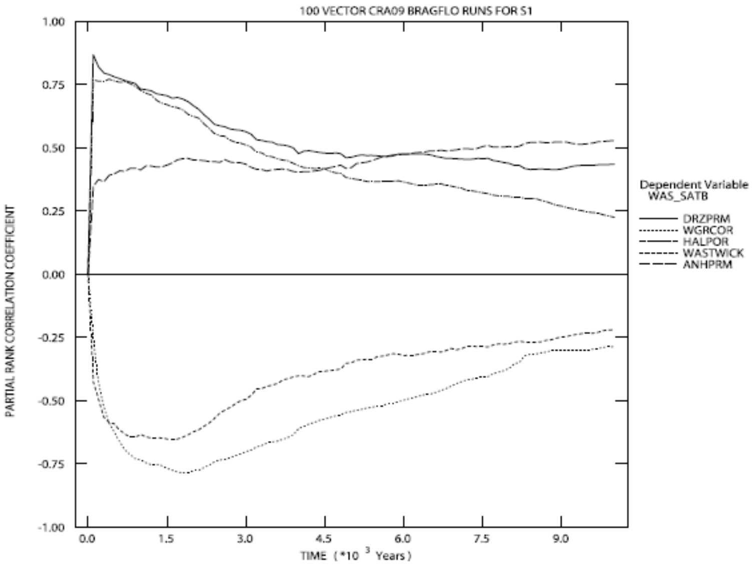
transport waste into and up a borehole. Confidence in the safety of WIPP comes not just from the demonstration that the permeability of the host rock is exceptionally low, but also the demonstration that the implications of the low permeability on all aspects of performance have been considered.

The Monte Carlo uncertainty analysis

As described previously, separating aleatory and epistemic uncertainty within a performance assessment allows the Monte Carlo uncertainty analysis to focus on the impacts of epistemic uncertainty. Statistical techniques, including regression analysis, can quantify the extent to which uncertainty associated with specific input parameters contributes to the range of model outcomes at both the component and system level. When applied to iterative performance assessments, Monte Carlo uncertainty analyses allow identification of those epistemic uncertainties for which the collection of additional data (potentially reducing uncertainty in the input parameter values) has the potential for reducing uncertainty in the overall estimates of disposal system performance. Equally importantly, the same analyses can identify those uncertainties that have little or no impact on the understanding of overall performance, and for which the collection of additional data is unlikely to either change estimates of mean system performance or reduce uncertainty in them. Thus, iterative performance assessments using rigorous approaches to Monte Carlo uncertainty analyses can provide a powerful tool to help direct research and development activities during characterisation of potential repository sites and development of the disposal system design.

In the example given earlier, regression analysis quantifies the qualitative insights about the effects of competing coupled processes. Figure 4 shows strong positive correlations between uncertainty in brine saturation in the disposal region and uncertainty in the permeability of the disturbed rock zone (DZPRM) and the anhydrite layers (ANHPRM), because higher values of both allow greater brine inflow into the disposal region. The figure also shows a positive correlation with uncertainty in the porosity of the intact halite (HALPOR), because this porosity forms the reservoir in the model from which brine may flow. Strong negative correlations appear with uncertainty in both the corrosion gas generation rate (WGRCOR) and a parameter used to characterise the rate at which brine is wicked up into the waste by capillary action (WASTWICK), because higher values of both parameters contribute to more rapid consumption of brine and a corresponding decrease in saturation (US DOE 2009, Section PA-7.1).

Figure 4: Partial rank correlation coefficients relating uncertainty in selected parameters to brine saturation in the waste disposal region at WIPP, 100 realisations of undisturbed performance



Source: US DOE, 2009, Figure PA-46.

Maintaining a clear separation of aleatory and epistemic uncertainty throughout the analysis allows recognition of the impact of uncertainty on both individual scenarios and overall performance. In the example shown above from the WIPP performance assessment, uncertainty associated with parameters that affect brine saturation and pressure in the disposal region is important with respect to releases due to spalling processes during drilling intrusion (waste carried into a borehole during rapid depressurisation), but can be shown to be of secondary importance to uncertainty associated with overall releases during intrusion. The largest contributor to total release from WIPP is mechanical erosion of waste into the borehole during drilling intrusion, for which uncertainty in the shear strength of the highly heterogeneous waste is the primary factor. Uncertainty regarding the solubility of the actinide elements in brine is a secondary contributor to overall uncertainty in estimates of long-term WIPP performance, influencing uncertainty in releases due to transport of contaminated brine to the surface during drilling intrusions (US DOE, 2008, Section PA-9.5).

Conclusions

Examples from repository analyses conducted by US DOE over the last two decades illustrate ways in which quantitative performance assessments can make important contributions to the overall safety case. Specifically, careful distinctions between aleatory and epistemic uncertainty in the design of large and complex analyses provide a rigorous foundation for subsequent uncertainty analyses, and allow the display of model outcomes both for individual scenarios of interest and for overall performance. Second, well-designed analyses permit evaluation of the performance of individual barriers within the context of the overall disposal system behaviour, revealing the importance of coupling among the components to provide isolation over the full range of potential conditions. Third, Monte Carlo uncertainty analyses permit quantification of the extent to which uncertainty in specific model input parameters impacts the distribution of model outcomes, both at the component and full-system level. In summary, a well-designed performance assessment does far more than simply evaluate compliance with a quantitative regulatory standard. It provides insight into what drives uncertainty in that estimate, it can provide guidance to research and development activities seeking to reduce uncertainty, and it can provide valuable confidence in the overall understanding of the long-term behaviour of the disposal system.

Acknowledgements

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The role of KURT and A-KRS in the development of generic safety cases in Korea

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Introduction

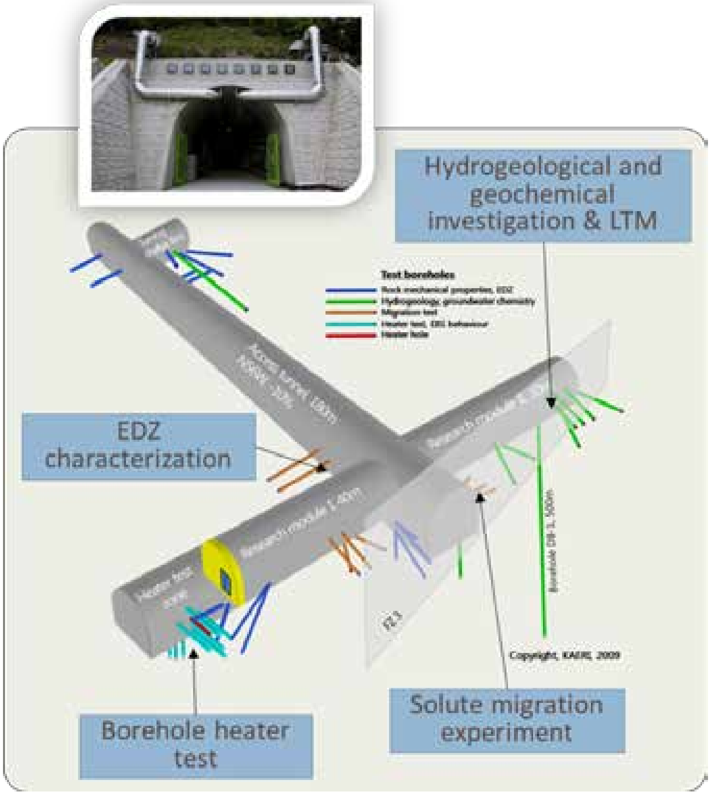
According to the draft guidelines for a deep geological disposal system for high-level wastes in Korea, the total annual risk for the average person resulting from the radiation exposure should not exceed 1.0×10^{-6}/yr and the expected radiation exposure to the average person for each scenario should not exceed 10 mSv/yr (NSSC, 2012). Regulatory compliance should be supported by multiple lines of reasoning such as probabilistic analysis of exposure dose and risk, uncertainty analysis, natural analogue, complementary safety indicators such as radionuclide concentration and release rates, secure of defence-in-depth. The integrated safety assessment should also be made and updated consistently based on up-to-date data and information for each stage of deep geological disposal of high-level waste (HLW) such as basic studies, site characterisation, design, construction, operation, closure, environmental monitoring after closure and so on. These are the bases for the development of a safety case in Korea. The Korea Atomic Energy Research Institute (KAERI) is now developing generic safety cases based on the Advanced Korean Reference Repository System (A-KRS) (Choi, 2011) and the KAERI Underground Research Tunnel (KURT) (Koh, 2011). The A-KRS is a geological disposal system for radioactive wastes from the pyroprocessing of PWR spent nuclear fuels in Korea. KURT is a small-scale underground research laboratory constructed and operated to assess the feasibility, safety, appropriateness and stability of the disposal concept by making various *in situ* tests and experiments. In this paper, the concept and long-term safety assessment of the A-KRS design, the role of the KURT in the development of safety cases, and the development of complex scenarios to support the development of generic safety cases in Korea are described.

The characteristics and role of KURT

KURT is a purpose-built generic underground research laboratory for the development of technologies for the siting and geological investigation of a geological repository of HLW. In addition, a number of researches for identifying the geological, hydrogeological and geochemical characteristics were implemented at the KURT facility. It is also being used to perform various *in situ* tests and experiments which can be used to develop safe and reliable disposal techniques for high-level radioactive waste in a deep geological formation.

A set of site characterisation and detailed design activities for the construction of KURT was completed in 2004, and construction was completed in November 2006. The host rock is granite, which is considered as a potential host rock type for a HLW disposal repository in Korea. The utilisation of radioactive material in KURT is not allowed. KURT has a total length of 255 m with a 180-m-long access tunnel and two research modules with a total length of 75 m. The maximum depth of the tunnel is 90-100 m from the peak of a mountain located above the site. The horseshoe-shaped tunnel section is 6 m wide and 6 m high, as shown in Figure 1.

Figure 1: Layout of the KURT facilities



Before constructing KURT, KAERI carried out geoscientific research at the KURT site as of 1997. Site investigation activities mainly consisted of the study of the baseline geological, hydrogeological, geochemical and rock mechanical properties of the host rock. These geoscientific studies were performed to improve the geoscientific investigation technologies for site characterisation, to provide geoscientific data for system development and safety assessment studies, and to develop a detailed site descriptive model at the field-block scale. The geology of the KURT site was also investigated by using the regional geological characteristics around the KURT site and the local geological characteristics of the site based on the geological data obtained from the construction of KURT and the drilled boreholes. The geology of the surrounding area of KURT was divided into the upper soil, a weathered layer, a low-angle fracture zone, a fracture zone and bedrock. Based on the results of a geological survey of the surface and a borehole investigation, these characteristics are investigated. A hydrogeological model that indicates the flow pattern of the groundwater in an objective medium was built based on the geological model and the hydrogeological characteristics of the geological components. For the investigation of the geochemical characteristics of KURT, rock-forming minerals and fracture filling minerals were identified using an optical microscopic study. Groundwater chemistry was also investigated using groundwater samples collected at various depths of a total of 17 boreholes at the KURT site. A groundwater flow modelling was performed at regional and local scales to understand the groundwater flow system at the KURT site. A regional-scale groundwater flow modelling was performed to understand conceptually the groundwater flow at the regional scale around KURT, and to derive a simulation domain and boundary conditions for local-scale groundwater flow modelling. The local-scale groundwater flow modelling was performed to understand the local-scale groundwater flow system around KURT and to introduce the boundary conditions for a site-scale groundwater flow and solute transport modelling around KURT. The distribution of lineaments and fracture zones at 500 m depth from the surface where a disposal repository would be expected to be located was estimated from the data on the lineaments and fracture zones by the analysis of the geological structural model and the groundwater flow model. From the estimated distribution of the lineaments and fracture zones, the study area for a disposal repository which is used as the layout of the A-KRS was determined based on the following conditions: *i)* fracture zone must not cross the repository; *ii)* the repository must stay away from the fracture zones greater than 50 m. After the study area was selected based on information on the fracture zones and the results of the groundwater flow simulation at the KURT site, a particle-tracking simulation was performed. The results of the analysis of the particle-tracking simulation allowed to determine the average groundwater flow rate in the fractured rock and fracture zone, the average transport distance from the hypothetical repository to the fracture zone, and the migration distance within the fracture zone, and this information was applied in the safety assessment of the A-KRS.

After the construction of KURT, several representative *in situ* tests and experiments were made to investigate the performance of repository barriers; a single hole heater test, an excavation damage zone study, a study on the characteristics of a solute migration in rock fractures, a hydrogeological and geochemical study. As KURT was approved as a "general facility", where no radioactive materials can be used inside, it is not allowed to conduct any experiments using radioactive nuclides in the facility. In 2011, KURT was used to investigate the influence of groundwater pressure on the fracture aperture size which controls the fracture transmissivity, at the request of Sandia National Laboratories (USA). Additional research, i.e. a spontaneous potential experiment, is in progress to determine the hydraulic properties and behaviour of the 3-D subsurface volume of the saturated fractured rock using a hydraulic head and streaming potential data. KURT was designated as a partner of the IAEA Underground Research Facility (URF) Network in August 2012.

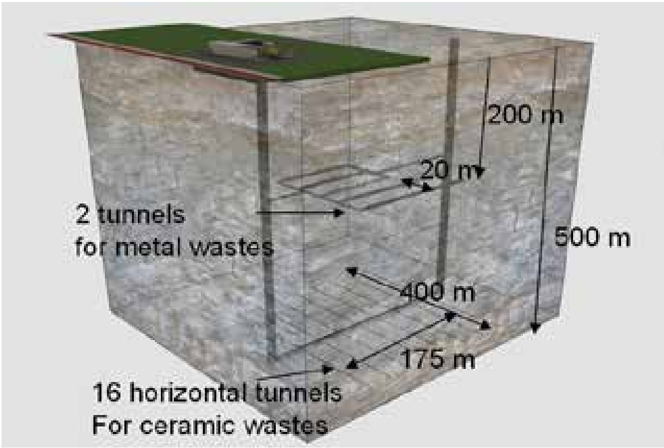
Data collected from geological investigations at KURT were used in the design and safety assessment of the A-KRS. KURT has also played a significant role in developing and demonstrating a repository system as well as the technologies needed for its construction and closure. The research experience gained at KURT has provided important information to validate the safety and feasibility of a disposal system and has made important contributions toward the successful implementation of a future commercial geological repository programme. The experience gained at KURT was further used in the safety assessment and the development of the safety cases for an intermediate- and low-level waste repository in Korea.

KAERI obtained government approval for a five-year R&D programme for a HLW long-term management system development in March 2012, which focuses on an enhancement of the performance of EBS of up to 20%, the establishment of the infrastructure for *in situ* demonstrations at KURT, and the development of safety cases based on the KURT environment. From 2012 to 2016, intensive experiments on the hydrogeological characterisation of MWCF and *in situ* long-term performance tests on the 1/3 scale engineered barrier system are major experimental research items to be executed at the KURT facility. The current dimensions of the research modules are limited, thus the KURT facility will be extended for the execution of the planned tests and experiments by 2014.

The design and safety assessment of the A-KRS

KAERI developed the A-KRS as a reference geological disposal system for HLW from the pyroprocessing of PWR spent nuclear fuels in Korea. The A-KRS is a conceptual hybrid-type repository system, in which two kinds of pyroprocessed radioactive wastes, low-level metal wastes and ceramic high-level wastes, are to be disposed of through separate disposal strategies. The design of the disposal system is based on the following three fundamental principles: the multi-barrier principle, the defence-in-depth principle and the radiation protection principle. As conceptually depicted in Figure 2, the A-KRS is considered to be constructed at two different depths in a geological media: 200 m depth, at which a possible human intrusion is considered to be limited after closure, for the pyroprocessed metal wastes with lower or no decay heat producing nuclides, and 500 m depth, believed to be in the reducing condition for nuclides with a much higher radioactivity and heat generation rate.

Figure 2: A-KRS, a conceptual hybrid-type repository system



The EBS design, including a disposal canister and a buffer, is based on international standards, as regulations regarding HLW repository performance are not yet established in Korea. Even though this design work is a general approach, the geologic data obtained from the KURT site are used where appropriate, i.e. the KURT site is used as a reference site for the design work. The development of a geological repository is carried out using iterative procedures of the design and performance assessment or safety assessment. The amounts of HLW is estimated based on the material balance from the pyroprocessing. The decay heat and radioactivity of HLW are calculated from a reference spent nuclear fuel, which is planned to be pyroprocessed after 10 years of cooling.

Since the total amount of HLW from the pyroprocessing of PWR spent nuclear fuel is not large, the disposal tunnels can be described with one disposal panel. According to the classification of the fracture zones, the disposal panel should be placed such that it has a 50 m safety distance from order 2 fracture zones. The disposal zone in the research area is selected to avoid the order 2 fracture zones. The disposal depth for ceramic waste is determined at the level at which the geochemical condition is reduced in order to minimise the corrosion rates of the disposal canisters and the dissolution rates of the ceramic waste forms. The distance from the repository to human beings should be sufficient such that the radionuclides released might decay out during the migration. Also, an inadvertent human intrusion into the geological disposal system should be avoided. The disposal tunnels for ceramic waste are located at a 500 m level below the surface. The layouts of the disposal system are determined mainly through a thermal analysis. It was determined that the peak temperature in the buffer should not exceed 100°C. Two kinds of disposal systems, horizontal and vertical, are developed.

We developed a computer program using Goldsim (2006) to assess the safety and performance of the A-KRS. Figure 3 shows the near- and far-field areas around the A-KRS repository system illustrating a connection to the biosphere envisaged typically under the geographical circumstances in Korea (Jeong, 2011). To check the design feasibility of the A-KRS, we made a safety assessment of the A-KRS for a reference scenario and three "What if?" scenarios: a well intrusion scenario, an earthquake scenario and an initial defect of a waste canister. All the scenarios are derived from the FEP analysis and a general scenario development methodology. Figure 4 shows the annual dose rates from nuclides with the total exposure dose rate for farming exposure group for the reference scenario. As shown in the figure, the exposure dose rates farming exposure group are far below the safety goal (10 mSv/yr), and are even lower than the natural background rate (2.4 mSv/yr) in Korea.

Figure 3: Conceptual modelling domain for the safety assessment of the A-KRS

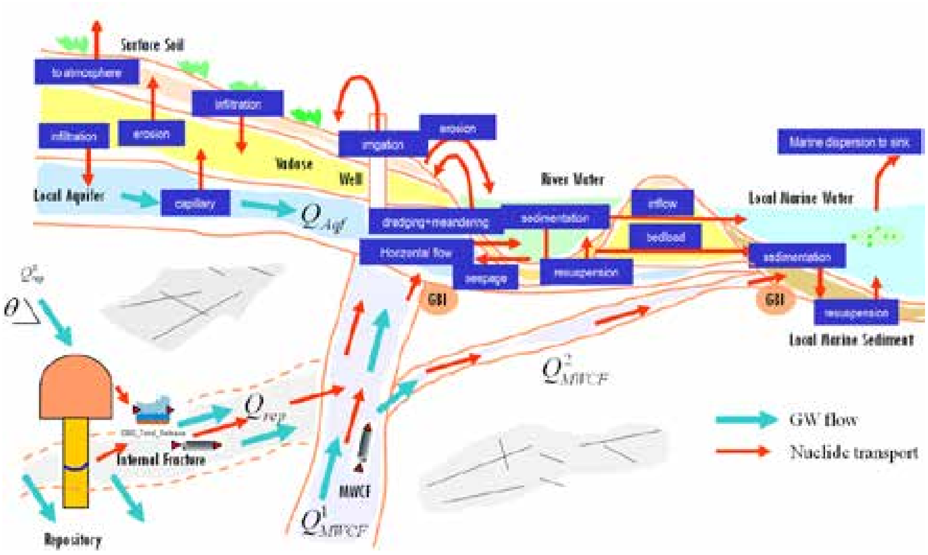
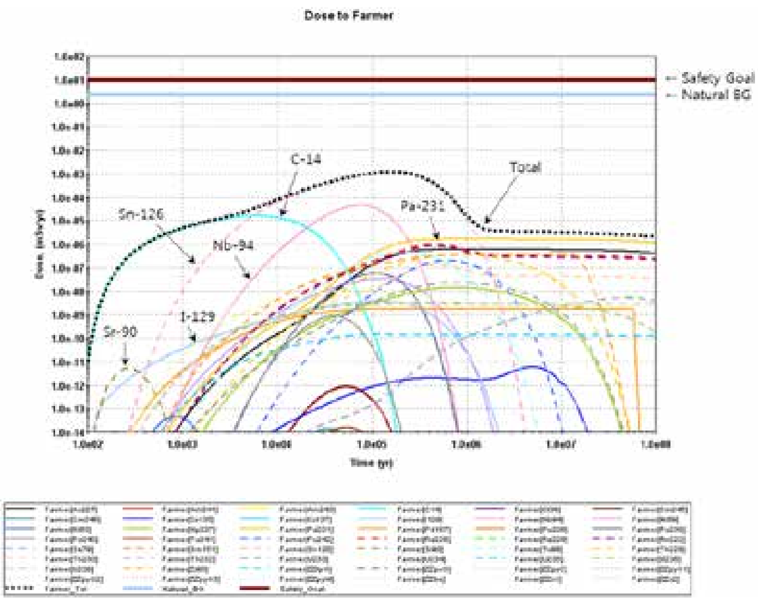


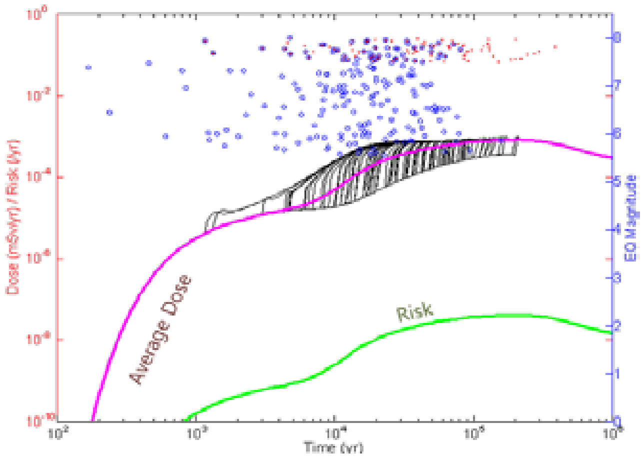
Figure 4: Dose exposure rate to the farming exposure group for reference scenario



Development of complex scenarios

According to the regulatory guidelines for a deep geological disposal system for the HLW in Korea, the total annual risk for the representative person resulting from radiation exposure should not exceed 1.0 ´ 10^{-6}/yr (NSSC, 2012). The use of complex scenarios considering the occurrence of each scenario for the risk-based safety assessment was also suggested, as was the use of complementary safety indicators such as radionuclide concentration and flux, natural analogue study and multiple source-term evaluation to support the development of safety cases. Therefore, we are developing methodologies for the development of complex scenarios, which are combinations of each scenario by considering the occurrence probability of each scenario to estimate a risk-based safety evaluation for integrating complex radiation exposure situations in the reference HLW repository system. We selected a combination of a reference scenario and an earthquake scenario as a reference complex scenario to develop methodologies for complex scenarios. The occurrence probability of an earthquake is expressed as a Poisson probability distribution function, the magnitude of the earthquake is expressed as a log-uniform probability distribution function and the hypocentral distance is expressed as a triangular probability distribution function. We consider two kinds of impacts of an earthquake on the repository system, a direct transport of radionuclides through the major water conducting features (MWCF) into the biosphere by the failure of internal fractures and the increase of groundwater flow rates from an earthquake. Based on these assumptions, we can obtain possible five complex scenarios. The Monte Carlo simulation results for the reference complex scenario are plotted in Figure 5. We found that the methodologies for the development of complex scenarios developed can be applicable to an estimation of a risk-based safety evaluation for integrating complex radiation exposure situations in a reference HLW repository system and can be used in the development of safety cases by making various risk profiles (Kim, 2012).

Figure 5: Monte Carlo simulation results for a reference complex scenario



Conclusion

KAERI is developing generic safety cases for a reference high-level waste repository in Korea based on the KURT environment. The A-KRS is a conceptual repository for the disposal of high-level wastes resulting from the pyroprocessing of PWR spent nuclear fuels in Korea. The layout of the A-KRS was determined through thermal analysis and the analysis of the geological structural model and the groundwater flow model of KURT, which is a purpose-built generic underground research laboratory for the development of technologies for siting and geological investigation of a geological repository of HLW. The host rock of the KURT site is granite, which is a candidate rock type for a HLW disposal repository in Korea. KURT is the only facility in Korea where radioactive waste disposal research can be conducted under repository conditions because a HLW disposal repository site has not yet been determined. Also, it is being used to perform various *in situ* tests and experiments which can be used to develop safe and reliable disposal techniques for high-level radioactive waste in a deep geological formation. The basic research conducted in KURT can be used in the development of generic safety cases for the HLW repository and has been used in the development of site-specific safety cases of low- and intermediate-level waste repository in Korea. Therefore, the research experience gained using KURT are providing important information to validate the safety and feasibility of the disposal system and making an important contribution to the successful implementation of a geological repository programme in the future. The long-term safety assessment of the A-KRS was performed using a program based on a GoldSim program. The design feasibility of the A-KRS was check by estimating the annual exposure dose rates for a reference scenario and three "What if?" scenarios. A methodology for developing complex scenarios by a combination of a reference scenario and an earthquake scenario was also created. This can be used to make a risk-based safety evaluation of the HLW repository system and the development of safety cases by making various risk profiles.

Acknowledgements

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Session 3.2

Safety Case for Preparing and Performing a Site Selection Process: The Swiss Example and Experience

The role of safety analyses in site selection: Nagra's experience from the ongoing Swiss site selection process

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Introduction and overview

Site selection in Switzerland takes place according to the so-called Sectoral Plan (BFE, 2008).[1] The Sectoral Plan defines three stages for site selection, the first of which was completed in 2011. For Stage 1 Nagra proposed in 2008 three siting regions for the HLW repository and six siting regions for the L/ILW repository (three of them overlapping with the siting regions for the HLW repository) (Nagra, 2008a). After a careful review by the responsible authorities and a broad consultation process, the Swiss Federal Council agreed in November 2011 to the siting regions proposed by Nagra. Currently Stage 2 of the Sectoral Plan is underway. Stage 2 has two broad aims: First, the selection of sites for the surface facilities within each siting region. This is done in close consultation with the siting regions through a participatory process. For that purpose, Nagra suggested 20 sites in the six regions early in 2012 (Nagra, 2011) and since then, extensive interactions with the siting regions have been ongoing. Second, in Stage 2 of the Sectoral Plan, the number of siting regions should be narrowed down to at least two for each repository type. The narrowing down is based on safety arguments and requires quantitative analyses (dose calculations according to detailed prescriptions by the authorities) and a qualitative evaluation (rating) of the siting regions according to criteria related to safety and technical feasibility defined in the Sectoral Plan.[2] Furthermore, a qualitative comparison of the siting regions is performed based on a limited number of important safety-related features (so-called "decision-relevant features") to assess whether some of the siting regions have clear disadvantages when compared with the others. In Stage 3 of the Sectoral Plan, further site investigations (seismic measurements, boreholes, etc.) will be performed in the remaining siting regions. The sites for preparing the general licences will then be selected and the corresponding projects (geological synthesis, facility design, safety analyses, etc.) and the licence documentation will be developed. Finally, the general licence applications will be submitted. The general licence will be followed by a construction licence and an operation licence, each of them building upon the necessary project work.

1. In Switzerland, all radioactive wastes must be disposed of in geological repositories. Two repositories are foreseen, one for L/ILW and one for SF/HLW/ILW. If the geological conditions allow, it is also possible to implement both repositories at the same site ("combined repository").
2. The same criteria were already used in Stage 1 to identify and select the proposed siting regions.

The importance of a sound scientific-technological basis for starting the site selection process

In the late seventies of the last century, an amendment to the Swiss nuclear energy law was issued that required the formal demonstration of the feasibility of safe disposal of all radioactive waste arising in Switzerland as a prerequisite for the continued operation of the existing NPPs and the possible construction of any new NPPs. This demonstration of feasibility had to be based on real data from a model site. Thus, a comprehensive geological programme was started at that time. The culmination of the extensive investigation and studies for the demonstration of disposal feasibility for SF/HLW/ILW was "Project Opalinus Clay", finalised in 2002 (Nagra, 2002a, 2002b, 2002c) which – after an extensive review (including a review organised through NEA) – was accepted by the Swiss government in 2006.[3] In the course of the review process, Nagra was also requested to make an assessment of the siting possibilities for the HLW repository as an input to the government decision (Nagra, 2005).[4]

Due to the complexity of Swiss geology, for the demonstration of disposal feasibility it was first necessary to perform complementary regional field investigations (e.g. seismic lines and boreholes, started in the early 80s) and then more detailed investigations at a model site for Opalinus Clay in northern Switzerland (borehole, 3-D seismics). Together with the comprehensive geological data set from hydrocarbon exploration and other sources as well as investigations in the two Swiss URL (Grimsel, Mont Terri) the information provided an excellent basis to start site selection according to the Sectoral Plan. Besides the regional and local geological data, this early phase of the Swiss waste management programme provided a wealth of information (including data from URL on host rock performance but also on the system of engineered barriers and other issues) and allowed Nagra to build up broad experience (team, models/tools, information and data) and projects in all relevant areas (waste inventory, geology, repository design, post-closure and operational safety, etc.). Thus, due to this very extensive programme, Nagra was in a good position to start the site selection process and could draw upon much experience and a broad information base.

Safety analyses to support the screening of Switzerland to identify siting regions (Stage 1 of the Sectoral Plan)

Stage 1 of the Sectoral Plan started with a "white map" of Switzerland; thus, the whole of Switzerland had to be evaluated with respect to siting possibilities for the two types of repositories. As a basis for the site selection process, the Sectoral Plan defines 13 broad criteria related to safety and technical feasibility grouped into four broad categories (see Table 1). To apply these criteria in Stage 1 of the Sectoral Plan, it was first necessary to develop a set of indicators that integrates more detailed and diversified information to inform each of these criteria. A total of 49 indicators were used in Stage 1 of the Sectoral Plan. The development of these indicators (including development of minimum and enhanced requirements, quantitative scales for evaluation of options, etc.) was based on safety considerations that included quantitative analyses (Nagra, 2008b).

The starting point of the quantitative safety considerations involved the definition of the barrier and safety concepts (including quantification of barrier efficiency) for both the L/ILW and the HLW repository (the latter with co-disposal of spent fuel, vitrified HLW

3. The demonstration of disposal feasibility for the L/ILW repository was accepted by the Swiss government in 1988, based on a project for the model site of "Oberbauenstock".
4. For the L/ILW repository, the site selection programme was started much earlier, leading to the selection of the Wellenberg site for the preparation of the general license application. This project, however, was abandoned in 2002 after the local population rejected the project.

from reprocessing and long-lived ILW)[5] and the allocation of the different waste sorts (approximately 120 different sorts as characterised in the Swiss radioactive materials and waste model inventory) (Nagra, 2008c) to the two repository types (Nagra, 2008b). For this purpose quantitative safety calculations were performed, looking at the contribution of the individual waste sorts to overall dose for different geological situations. These calculations involved normal release situations (for a range of different generic host rocks) but also the very long-term evolution with the eventual denudation of the waste emplacement rooms by erosion. The issue of erosion was addressed specifically to assess the time period during which the geological setting has to ensure a minimum amount of protection. Using simplified model concepts, the time scales of relevance were defined to be 100 000 years for the L/ILW repository and one million years for the HLW repository.[6]

The quantitative analyses involved not only dose calculations to derive requirements especially on host rock properties (minimum thickness, requirements on hydraulic conductivity/transmissivity) but also system analyses[7] addressing other performance indicators mainly related to long-term geological stability (e.g. the role of and requirements related to uplift and erosion) and related to perturbing phenomena [the role and importance of excavation damage (Poller, *et al.*, 2013; Nagra, 2013), of gas generation (Nagra, 2004, 2008d), of pH-plume and of heat output by HLW/SF]. These model calculations were complemented by semi-quantitative considerations e.g. for radionuclide retention in the host rocks (mineralogy, pore space, geochemical conditions, etc.). Furthermore, criteria related to engineering feasibility were also developed (rock strength, maximum depth, geometrical aspects, etc.). Based on these analyses and considerations it was possible to develop indicators addressing large-scale issues such as long-term evolution (e.g. uplift of the Alps) and geometrical issues at a more localised regional scale (depth of rock layers, large-scale regional fracture zones, over-deepened valleys, necessary size of host rock blocks, etc.).

Based on the indicators, the proposed siting regions were developed in a stepwise manner, starting with screening to identify large-scale suitable tectonic situations (considerations mainly based upon uplift/erosion, deformation patterns), followed by identifying potentially suitable host rocks within these large-scale areas (considerations mainly based upon transport properties) and finally to find suitable configurations of the host rocks within the large-scale tectonically suitable areas. Based on a qualitative evaluation of these configurations, the siting regions were identified. For the siting regions identified, no dose calculations were required according to the Sectoral Plan.

In preparation for Stage 2, it was necessary to formally assess the quality of information available and to evaluate the importance of existing uncertainties for the more quantitative assessments in Stage 2. For that purpose Nagra developed a broad set of test calculations which showed that for all proposed siting regions the calculated doses are well below the regulatory safety criterion (Nagra, 2010). This is a clear indicator that the approach chosen and the requirements formulated in Stage 1 are well suited to identify siting regions with properties that allow the implementation of safe repositories.

5. The proposed safety and design concepts for the two repositories draw strongly upon the experience available from Nagra's long RD&D programme. For Stage 1 a modular system of the repositories was developed that allowed easy adaption to the specific conditions encountered at each of the sites.
6. These time scales were accepted by the Swiss safety authorities (ENSI, 2010a) in their review and are now also a boundary condition for the work in Stage 2.
7. Some of these analyses were already available from earlier studies.

Safety analyses to support the narrowing down of siting regions (Stage 2 of the Sectoral Plan)

As mentioned in the introduction, in Stage 2 of the Sectoral Plan quantitative dose calculations and a qualitative evaluation of each of the siting regions are required. These are complemented with a comparison of the siting regions with respect to a limited number of key features to check if "clear disadvantages" exist for any of the siting regions.

The dose calculations are developed and performed according to detailed prescriptions by the Sectoral Plan (BFE, 2008) and the safety authorities (ENSI, 2010b). For that purpose, first an evaluation is made of the processes and parameters important to safety, then the uncertainties for these processes and parameters are assessed and based on this evaluation the cases to be considered in the dose calculations are defined.[8] As mentioned above, this was done by Nagra in a preliminary manner in preparing for Stage 2 of the Sectoral Plan (Nagra, 2010). Currently, this work is being updated.

For the qualitative evaluation of the different siting regions in Stage 2, the same 13 criteria are used for the narrowing-down process and most of the indicators used in Stage 1 will also be used in Stage 2, although for some of them small refinements are necessary due to the changed scale of consideration (Stage 1 looked at the whole of Switzerland at the scale of larger regions whereas Stage 2 looks at the smaller regions defined in Stage 1, with a more localised focus) and a very few indicators of Stage 1 are no longer applicable in Stage 2 due to the different spatial scale.

Finally, as an important part of Stage 2, the qualitative comparison of the siting regions for a limited number of decision-relevant features will be very important. The evaluation of these decision-relevant features builds on the evaluation of a limited number of corresponding indicators already used for the general evaluation. It is expected that this comparison will lead to the identification of "clear disadvantages" for some of the siting regions, which will then not be further considered in Stage 3 of the Sectoral Plan.

Safety analyses for the general licence application (Stage 3 of Sectoral Plan)

The endpoint of Stage 3 is the general licence application that will require a safety case (post-closure safety) expected to be comparable in its nature to the safety case for project Opalinus Clay. This will involve all elements of a full safety analysis [phenomenological analyses, scenario analysis, system analysis (looking at specific phenomena and the performance of specific components) and dose calculations both with deterministic and probabilistic analyses]. The general licence application will also include a preliminary analysis of operational safety.

Summary and conclusions

Nagra's experience from the ongoing Swiss site selection process, and Nagra's assessment of the role of safety analyses in site selection, can be summarised by the following broad points:

- Site selection in Switzerland takes place according to the Sectoral Plan which was developed with broad consultation involving the different interest groups. Three stages are used for site selection, each of them including a detailed technical review by the relevant authorities, broad consultation with all interest groups leading to a decision by the Swiss Federal Council. This broadly based decision-making process

8. However, the type of cases to be considered is restricted to a limited number of situations as defined by the safety authorities (ENSI, 2010b).

that includes many presentations and discussions with specialist and non-specialist stakeholders is also expected to lead to a better understanding of the "why here and not there" by the public – an issue considered important in gaining public acceptance.

- In each of the three stages, safety (and ensuring technical feasibility) is dominating the site selection process. Due to the change in focus and the increasing level of detail, the nature of safety analyses changes in the different stages going from generic analyses in Stage 1 to a detailed safety case for each repository type in Stage 3.
- The available information and experience from approximately 30 years of research and development and from preparing earlier milestone reports is essential in the site selection process and includes information on the waste inventory, geology, repository design, post-closure and operational safety.

Table 1: Criteria for site evaluation from the viewpoint of safety and technical feasibility

Criteria group	Criteria
1. Properties of the host rock and the effective containment zone	1.1 Spatial extent 1.2 Hydraulic barrier effect 1.3 Geochemical conditions 1.4 Release pathways
2. Long-term stability	2.1 Stability of the site and rock properties 2.2 Erosion 2.3 Repository-induced influences 2.4 Conflicts of use
3. Reliability of geological findings	3.1 Ease of characterisation of the rock 3.2 Explorability of spatial conditions 3.3 Predictability of long-term changes
4. Engineering suitability	4.1 Rock mechanical properties and conditions 4.2 Underground access and drainage

Source: BFE, 2008 (English translation).

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The regulatory perspective: Role of regulatory review of the safety case for preparing and performing the Swiss site selection process

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Overview of the site selection process and the realisation steps of repository for radioactive waste in Switzerland

Swiss legislation requires for all types of radioactive waste the safe and permanent disposal in deep geological repositories within Switzerland by stipulating a step-by-step procedure for site selection and licensing of the disposal facilities. Each step requires safety considerations or safety analyses which are reviewed by the Swiss Federal Nuclear Safety Inspectorate (ENSI, former HSK). The principle steps of radioactive waste disposal include:

i) Demonstration of disposal feasibility for all types of radioactive waste in Switzerland.

ii) Site selection process in three stages to narrow down the number of suitable sites to one for realisation ("Sectoral Plan for Geological Repositories").

iii) Construction, operation and closure of the repositories according to Swiss legislation in five steps [licence for geological investigations, general licence (decision-in-principle), construction licence, operation licence, closure order, Figure 1].

Step i) was completed in 2006 when the Swiss Federal Council (Swiss federal government) approved the demonstration of disposal feasibility for high-level and long-lived intermediate level waste submitted by the implementer in 2002.

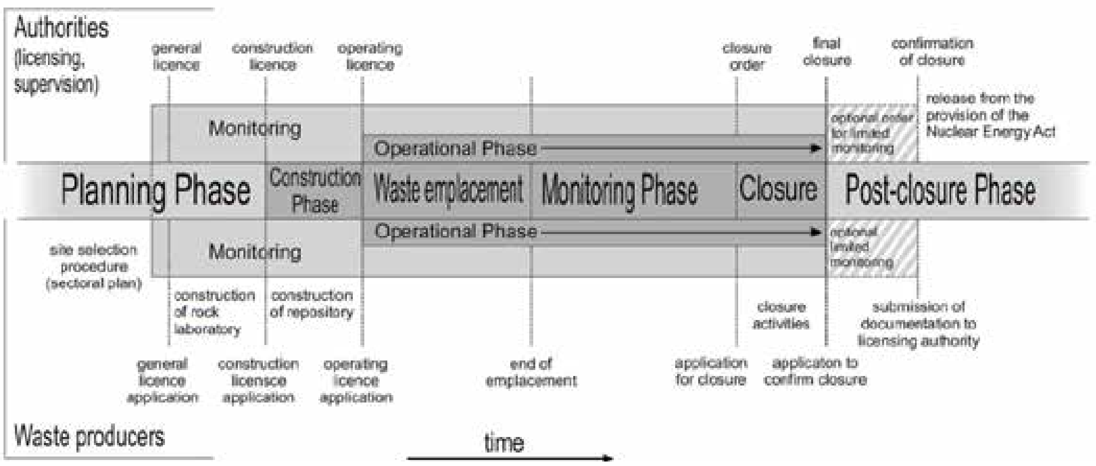
The site selection process started in 2008 [Step ii)] according to a site selection concept (BFE, 2008) which was approved by the Federal Council after broad consultation. The process is organised and co-ordinated by the Swiss Federal Office of Energy (SFOE, in German BFE) and is carried out in accordance with the land use planning legislation. Safety is of highest priority in the site selection process, but socioeconomic aspects are also taken into account. The site selection procedure is based on a stepwise approach. In the first of three stages, six potential siting areas for L/ILW and three potential siting areas for HLW were proposed by the implementer (Nagra) for each repository type, based on safety criteria defined by the regulatory authority ENSI. ENSI has reviewed the documentation and has agreed with the proposed geological siting areas (ENSI 33/070). Agreement has further been supported through confirming reviews by the Federal Nuclear Safety Commission (CNS), the cantonal experts (AG SiKa/KES) and a German expert group (ESchT). A three month broad public consultation procedure was undertaken at the end of 2010. The Federal Council finally approved the potential siting regions by the end of 2011, ending thus Stage 1 of the site selection process.

In Stage 2, which is currently ongoing, these potential repository siting areas will be compared among each other on the basis of provisional safety assessments, a qualitative evaluation of each siting area based on safety criteria and a standardised intercomparison procedure. The proposal by Nagra will again be reviewed by ENSI and other expert groups including the CNS, the cantonal experts and experts from the neighbouring countries. Stage 2 is expected to be concluded by federal government approval in 2016.

The remaining siting areas – at least two for each repository type – will be investigated in detail during Stage 3 of the process. Nagra will propose the sites for each repository type and apply for a general licence for this site. The safety case will include a full safety assessment for each selected site as well. Based on the results of this process, a repository site will be selected for each repository type and approved by the federal government. The general licence will be granted by the Federal Council and must be approved by Parliament. The approval is subject to a facultative referendum (national vote).

This paper describes the role of the ENSI, the safety requirements for the site selection process, presents examples of regulatory review and elucidates its feedback to the site selection process.

Figure 1: Simplified schematic representation of the processes involved in planning, construction, operation and closing a deep geological repository (ENSI, 2009)



The role of ENSI in the site selection process

The Swiss Federal Office of Energy has the leading role in the site selection procedure, while the Swiss National Co-operative for Radioactive Waste Disposal (Nagra) is responsible for the preparation of proposals and licence applications on behalf of the waste producers. The proposals by Nagra are reviewed from a technical point of view by the ENSI and other authorities and commissions [Nuclear Safety Commission, a cantonal expert group (AG SiKa/KES), and an expert group (EGT, formerly KNE (Commission on Nuclear Waste Management) supporting ENSI]. ENSI bears overall responsibility for the safety assessment of geological siting areas and sites and is responsible for the specification of safety requirements for the site selection process. In addition, ENSI leads the Technical Forum on Safety, co-ordinates its work and heads the secretariat. The main function of the Technical Forum on Safety is to discuss and answer technical and scientific questions on safety and geology from the public, the communes, the siting regions, organisations, cantons and public bodies of affected neighbouring countries.

Selection of geological siting areas for L/ILW and HLW (Stage 1)

Safety criteria for the selection of potential siting areas

The aim of Stage 1 was to identify several geological siting areas for L/ILW and HLW. Therefore, 13 criteria relating to safety and technical feasibility (Figure 2) were defined by ENSI based on international documents and on the experience from past reviews of the feasibility studies and the requirements of the Swiss legislation (HSK, 2007, Rahn, *et al.*, 2008): The long-term stability and the retardation properties of the host rock and the geosphere are paramount qualities which have to be considered in the site selection process for a deep geological repository.

Figure 2: Criteria for site evaluation from the viewpoint of safety and technical feasibility (BFE, 2008)

Criteria group	Criteria
1. Properties of the host rock and the effective containment zone	1.1 Spatial extent 1.2 Hydraulic barrier effect 1.3 Geochemical conditions 1.4 Release pathways
2. Long-term stability	2.1 Stability of the site and rock properties 2.2 Erosion 2.3 Repository-induced influences 2.4 Conflicts of use
3. Reliability of geological findings	3.1 Ease of characterisation of the rock 3.2 Explorability of spatial conditions 3.3 Predictability of long-term changes
4. Engineering suitability	4.1 Rock mechanical properties and conditions 4.2 Underground access and drainage

The safety criteria of the Sectoral Plan are of descriptive nature. One reason is that ENSI guideline ENSI-G03 states that radiological consequences from a repository must not exceed a yearly individual dose of 0.1 mSv, but does not define by what means this safety criterion has to be met. By choice of an adequate combination of natural and technical barriers, the implementer must comply with the safety criterion. ENSI has therefore chosen not to set limiting values for parameters such as e.g. thickness of the host rock layer or hydraulic conductivities. In contrast, the implementer had to define the inventory of each repository type and then derive a set of quantitative requirements for potential host rocks for the planned repositories (e.g. dimensions of the repositories, thickness and hydraulic conductivities of host rocks). Based on these requirements, the implementer had to identify suitable large-scale geological areas for hosting a repository. Next, potentially suitable host rock formations within these areas had to be identified by excluding regions with e.g. enhanced tectonic complexity or probable occurrence of deeply eroded glacial valleys.

Implementation of the regulatory review

On behalf of the waste producers, Nagra proposed potential siting areas based on the criteria as shown in Figure 2. The review of ENSI was based on this proposal of geological siting areas, its argumentation and qualitative assessment of the safety criteria. The focus on the review relied on the following safety relevant key questions:

- Are the implementer's requirements for the choice of the host rock formations/ geosphere correctly derived and sufficient to comply with the dose limit of 0.1 mSv/year?
- Did the implementer use all relevant geological information, and is the quality of the data adequate for the choice of the site regions?
- Did the implementer take the 13 criteria into account for the choice of the siting regions?
- Is the implementer's procedure to choose these regions transparent and plausible?

To answer these key questions, ENSI:

- carried out selected model calculations based on independent modelling codes (e.g. estimation of radionuclide release, migration across the barrier system to the biosphere, dose calculations, gas production and gas transport within a repository) to verify the allocating of the waste to the L/ILW and HLW repositories and the specifications for the quantitative safety requirements for the host rock by Nagra;
- and KNE organised workshops on scientific topics such as glacial erosion and earthquake/neotectonic activity with national and international experts to evaluate the current scientific knowledge;
- started research projects to clarify scientific issues on topics such as deep glacial erosion and long-term climate change (encompassing far beyond 10 000 years).
- was supported by external experts for specific topics such as e.g. rock mechanical properties, glacial erosion, gas production rates in the repository and their effects on the host rocks, available geological information, Quartenary geology, GIS systems to derive the siting areas and by a national expert group (Commission on Nuclear Waste Management, KNE) which published a separate expert report (KNE, 2010).

Overview of review results

The review of ENSI concluded that the geological siting areas for L/ILW and for HLW proposed by Nagra are all suitable for further investigations within the next stage of the Sectoral Plan (ENSI, 2010b). The provided geological information and the data base of the safety considerations are presented in a clear manner and can be considered sufficient for the purpose of Stage 1. For the identification and narrowing down of suitable geological siting areas, the following safety issues have been considered:

- suitable depth;
- sufficient volume of host rock for locating a repository (thickness and lateral extent of the host rock, including extra space to be flexible in siting);
- distance to potentially active faults or faults subject to reactivation;
- distance to zones of elevated tectonic overprint;
- distance to zones of recent seismic activity;
- distance to over-deepened glacial valley filled with young deposits (glacial erosion).

Similar to Nagra, ENSI assessed qualitatively all safety criteria for all resulting siting areas in order to gain a general overview of the suitability of each area. For some cases, ENSI differed in its assessment from Nagra, e.g. with respect to the effects of the repository on the host rock (such as gas production and gas transport, sensitivity to thermal effects, thermal-hydraulic-mechanical coupled processes, formation of an excavation damaged zone around underground structures). In its assessment ENSI points out several critical issues with respect to clay and clay-rich host rocks. However, these differences in the assessments between Nagra and ENSI had no influence on the overall evaluation of the geological siting areas and the final result. At the same time, ENSI formulated recommendations which Nagra must consider in more detail in Stages 2 and 3, such as:

- The expected radioactive waste forms of decommissioning from national (PSI) and international research facilities (CERN) have to be characterised in more detail.
- For a repository depth of 650 to 900 m, it is likely that additional measures of tunnel stabilisation will be necessary. Accordingly, the geomechanical properties in such depths and its impact on long-term safety have to be investigated in more detail.
- The definition of a regional erosive basis within the siting areas should rely on a methodical approach which better incorporates the potential of lateral and vertical fluvial erosion and the interaction between fluvial and glacial erosion.

Lessons learned and feedback to the site selection process

In Switzerland only marine sediments such as the Helvetic Marls, Effinger Member Marls, middle and lower Jurassic clay-rich rocks (“Brauner Dogger” and Lower Jurassic = Opalinus Clay) have been proposed as suitable host rocks for a repository of L/ILW. All four proposed host rocks have favourable to very favourable retention capacities. Several other rocks containing significant amounts of clay minerals fail single criteria of safety and technical feasibility requirements (mainly due to heterogeneity, insufficient self-sealing capacity or difficulties to explore the occurrence with sufficient reliability).

For a HLW repository only Opalinus clay was considered as suitable host rock due to its superior homogeneity with respect to lateral and vertical clay content and the resulting self-sealing capacity, its low hydraulic conductivity and very favourable retention capacity of the host rock.

As a result, the quantitative comparison does not differentiate between a generic host rock with good sorption properties and a host rock with less favourable sorption properties due to the high retention capacities of the technical barriers. Therefore, the barrier effect has to be assessed for the multiple barrier system and for each single barrier in Stage 2 to show the robustness of the system. This may indicate which natural barrier of the siting areas has advantages and which natural barrier has disadvantages.

Since the site selection process will continue for more than ten years and the alternative is kept open to return to geological siting areas which have kept as secondary options as part of the narrowing-down process, the isolating rock zone (including the host rock) in all potential siting areas should be given special protection against drilling, open pit and underground mining and underground construction activities. Such a protection has been required with the approval of Stage 1 of the siting process. Protection is reached by a set of issued maps with contoured depth values, which define the maximum depth level that drillings are allowed to reach. These maps have been given to the cantons, which are responsible for co-ordinating land use planning, including approval of any proposals for drilling activity (mainly for geothermal purposes). The cantonal authorities apply the maps and inform ENSI about any conflicts.

For an efficient review process the documentation for all modelling calculations has to be transparent and comprehensive; an electronic folder of input parameters and results of the implementer are valuable. In addition, a separate report containing the questions of the regulator and the associated answers of the implementer has proven to be a valuable tool for transparent and efficient information transfer.

Selection of at least two siting areas each for L/ILW and HLW (Stage 2)

Safety requirements for the comparison of geological siting areas in Stage 2

In each of the identified siting areas, the implementer must choose at least one site for which a provisional safety analysis must be carried out. In addition to the safety assessment for the individual sites, a qualitative evaluation of the criteria with respect to safety and technical feasibility and an overall comparison of the sites are taken into account to find at least two siting areas for each repository. Sites can only be excluded if clear disadvantages as compared to the other site are identified in Stage 2. Siting areas excluded in Stage 2 are maintained as a secondary option until the general licence.

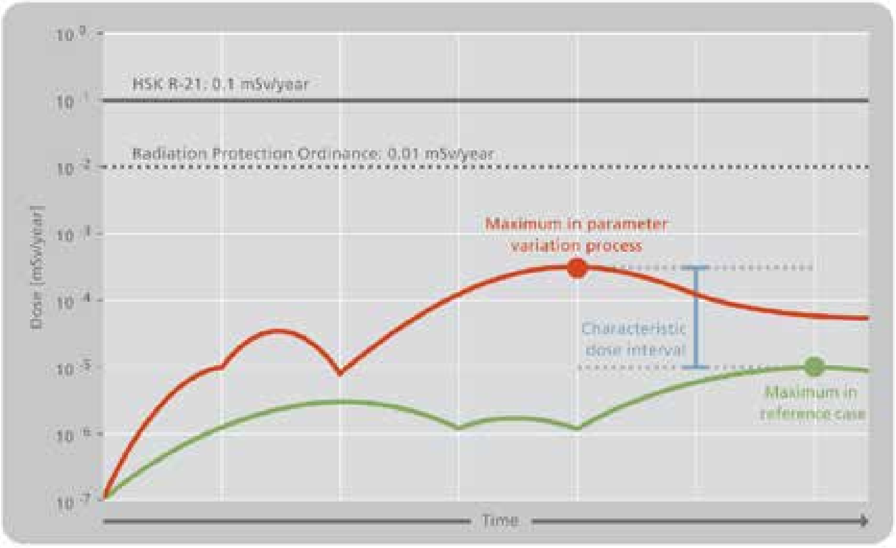
ENSI has specified the safety requirements on the provisional safety analyses and the comparison based on safety criteria (ENSI, 2010a):

- The choice of these siting areas must be transparent and plausible. In order to achieve this objective, the implementer has to show the contribution of the technical and natural barriers to overall safety and compliance with the safety criteria (annual dose of less than 0.1 mSv). Any provisional safety analysis must be based on adequate knowledge of the host rock properties, the local geosphere and the geochemical conditions.
- In a first step, the implementer must define a reference scenario for the provisional safety analysis which should correspond to a realistic description of the expected evolution of the repository including realistic assumptions for the development of the waste forms, the near field, the far field, the geosphere and radionuclide transport to the biosphere. Within this reference scenario, the implementer has to define a reference case and reference values for the transport-relevant parameters to calculate the expected radiological consequences of a potential repository at each chosen site.
- In a second step, the implementer must vary key parameters relevant for nuclide migration such as water flow, nuclide-specific diffusion parameters, geochemical parameters (solubility, sorption coefficients) according to a procedure to be defined by ENSI. The results of these calculations will provide a measure for the robustness of each site. For spent fuel elements, additional variations such as increased dissolution rates for the fuel elements or a limited canister life time will have to be taken into account.
- The sites will then be categorised according to the maximum annual dose obtained in the parameter variation procedure; the best category entails all sites with a maximal yearly dose of less than 0.01 mSv. The next category comprises sites with a maximum annual dose between 0.01 mSv and 0.1 mSv. Sites with a maximum annual dose higher than 0.1 mSv are excluded from further consideration (Figure 3).
- Remaining sites are then evaluated using the qualitative safety criteria (Figure 2). Only those sites that are designated at least "suitable" in the overall evaluation on a qualitative evaluation scale of suitability (e.g. very suitable, suitable, limited suitability, less suitable) will be further considered in Stage 3. In addition, sites can be excluded if clear disadvantages as compared to the other sites are identified at this stage.

In order to realise the objectives of the upcoming Stage 2, the state of knowledge of the geological information at the sites has to be sufficient to perform the provisional safety analyses, the qualitative evaluation and the comparison. Therefore, in preparation for Stage 2, the Sectoral Plan required Nagra to clarify with ENSI, at an early stage, the need for additional investigations aimed at providing input for the provisional safety analyses and the comparison. In order to document Nagra's technical-scientific assessment of this need Nagra submitted a technical report to ENSI in October 2010 (Nagra, 2010).

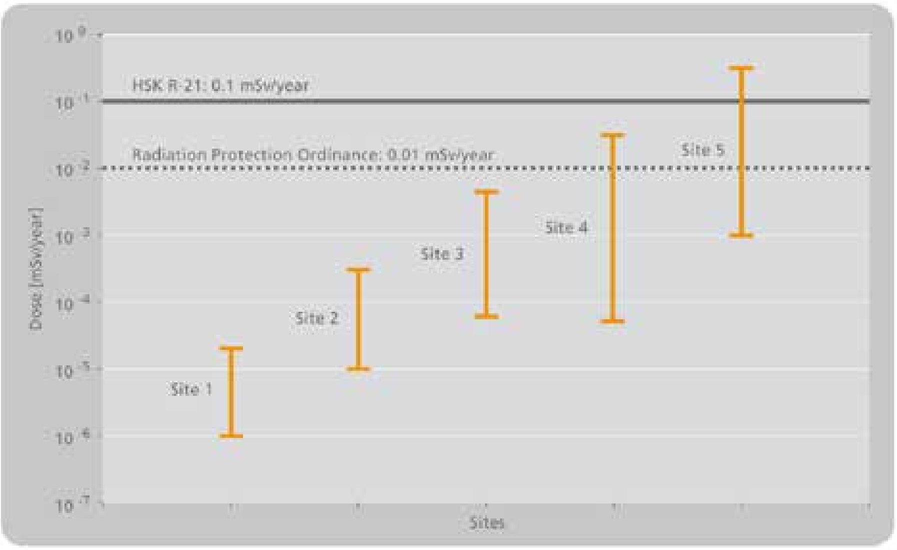
Figure 3: Method for comparing the numerical calculations (BFE, 2008)

(a) Determining the characteristic dose interval for a deep repository used in the comparative approach: the evolution with time of the calculated doses is calculated for the reference case (green) and for the cases defined using the parameter variations (red). The dose maxima in each case that define the dose interval are shown (filled circles). Note: The dose curves shown are hypothetical examples.



Note: The dose curves shown are hypothetical examples.

(b) Dose intervals from the provisional safety analyses for five hypothetical sites (that could be in different host rocks). Each site is compared with the radiologically best site (site with the lowest dose in the reference case, in this example Site 1). In this example, Site 5 is excluded since the upper value of the interval of dose maxima is above the ENS-G03 protection objective of 0.1 mSv/a. Sites 1, 2, 3 and 4 are suitable from the viewpoint of safety as their dose intervals lie below 0.1 mSv/a. Sites 1, 2 and 3 are also considered to be equivalent in terms of safety as their dose intervals lie below the threshold value of 0.01 mSv/a. Site 4 is excluded from further consideration as its dose interval does not overlap with that of the best site (Site 1) and goes beyond 0.01 mSv/a.



Overview of review results and lessons learned

ENSI and their experts have reviewed this report; the corresponding review was published in March 2011. The main findings of the review are that, together with the complementary investigations proposed by Nagra and requested by ENSI, the state of knowledge of the geological conditions at the sites will be sufficient to perform the provisional safety analyses, and that therefore ENSI is not requesting any investigations within Stage 2 that require a licence (e.g. the drilling of deep boreholes). At the same time ENSI required 41 safety issues such as:

- improved knowledge of properties for the new host rocks Effinger member Marls and "Brauner Dogger" for a L/ILW repository;
- detailed and systematic description of the flow paths in the siting areas;
- detailed investigation for rock mechanical properties (risk assessments for underground structures).

Therefore, ENSI has specified the requirements on engineering risk assessment for underground structures and safety considerations for underground access in a separate document (ENSI, 2013b).

This view was broadly confirmed by the other authority bodies. At the same time, it was requested that the cantonal experts and CNS carry out specific seminars, in which Nagra discusses the results of the complementary investigations proposed by Nagra and requested by ENSI. This process to review the additional geological information is described in ENSI 33/155 (2013a).

Another result of the reviews of the different expert groups was the request to specify the qualitative evaluation method based on safety criteria for Stage 2. Therefore, ENSI carried out several meetings with representatives from the different expert groups and discussed advantages and disadvantages for different multiple criteria analyses. As a result of the discussion important elements of the methods for comparison are described in ENSI 33/154 (2013c) (specifications of the safety method to select at least two geological siting areas for each repository in Stage 2):

- The method has to be carried out in a clear and transparent manner.
- The variations and existing uncertainties in data, processes and models and calculation of the resulting proposal of siting areas which will be investigated further in Stage 3 has to be analysed.
- Nagra must assess specific safety characteristics for optimisation (designated "decision-relevant features") to select at least two siting areas. For the evaluation of clear safety disadvantages, siting areas must be compared to other siting areas so as to proceed with the exclusion of sites in Stage 2.
- Criteria and indicators which are evaluated as less favourable have to be shown separately.

Based on an assessment of the remaining sites according to the safety criteria, the implementer will propose at least two sites per repository type. This proposal will be reviewed by ENSI. If approved by the Swiss government, the site selection enters the final stage (Stage 3) of the Sectoral Plan, in which the implementer collects detailed site-specific data at all the sites, prepares an updated safety assessment, chooses the sites for the realisation of the two repositories and applies for general licences. ENSI will review the safety assessments. The general licence application will be granted by the Swiss government, and must be approved by Parliament: Approval is subject to a facultative national referendum.

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Session 3.3

Developing the Safety Concept and the Design of a Geological Repository: The Belgian Example

RD&D steering of a geological disposal programme in poorly indurated clays

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For more than thirty years, Belgium has been investigating clay formations for its potential suitability to host a geological disposal. The R&D programme initiated as early as 1974 by the Belgian Nuclear Research Centre (SCK•CEN) at Mol was pursued from the early 1980s under ONDRAF/NIRAS' responsibility. These studies quickly focused on the Boom Clay formation at Mol-Dessel, in north-eastern Belgium, as a potential host formation for a geological repository.

The state of scientific and technical research on the possible disposal of high- and intermediate-level radioactive waste (B&C waste) in clay layers was presented in decennial safety assessment reports (ONDRAF/NIRAS, 1989, 2001). The national and international peer review of the second Safety and Feasibility interim report SAFIR 2 (NEA, 2003) acknowledged the maturity of the Belgian scientific programme and endorsed ONDRAF/NIRAS' conclusion to pursue the RD&D programme associated with a safe and feasible geological disposal in poorly indurated clays. Next to the continuing necessity of RD&D in all relevant areas of the Belgian programme, the NEA International Review Team (IRT) highlighted three main areas of activity that ONDRAF/NIRAS should strengthen to move on to the implementation phase. First, the IRT acknowledged the novel and innovative methodological concepts (i.e. safety functions, alternative safety indicators) introduced in its programme and recommended that ONDRAF/NIRAS move forward in this direction and improve the methodology for a more systematic, consistent and comprehensive treatment of uncertainties. Secondly, the IRT recommended further development of the EBS. Understanding of the engineered component's behaviour, its physic-chemical evolution with its inter-dependencies with the other components, its feasibility as well as its performance was indeed limited at the time of SAFIR 2. Last, the maturity of the geological programme suggested that it was time to initiate a dialogue with the regulators, policy makers and other interested stakeholders in Belgium on a number of aspects, including the regulatory framework and guidance on how to proceed with siting and with the associated decision making.

These issues pointed in the direction to urge ONDRAF/NIRAS to develop the frame and tools to embed its national programme in a stepwise decision-making plan with key milestones supported by successive safety cases stretching over the next decades towards the licensing process. The development of the first safety case of its kind (SFC1) led ONDRAF/NIRAs to re-orient the geological disposal programme towards a multifaceted development programme, bringing together technical, regulatory and societal aspects, each of these having their own weights and priorities.

To serve these ends, ONDRAF/NIRAS has reassessed the organisation of its geological disposal programme. In recent years ONDRAF/NIRAS has elaborated a safety and feasibility strategy to frame and formalise the stepwise and iterative development of geological disposal in a coherent and integrating manner, able to fulfil all identified management requirements (ONDRAF/NIRAS, 2009a, 2013). The strategy sets out in broad terms how it is envisaged that safe disposal will be achieved. It is mainly based on safety principles established by the IAEA which are, among others, passive safety, defence-in-depth, robustness, best available technology and optimisation. The safety and feasibility strategy also includes the boundary conditions to be met and the requirements to be satisfied. Boundary conditions include, for example, relevant international and national regulatory frameworks, institutional policy and conditions required by other stakeholders. Within this strategic frame and along the same safety principles, a safety assessment methodology was also developed for a comprehensive and consistent treatment of uncertainties in order to steer the RD&D (ONDRAF/NIRAS, 2009b, 2013).

The safety and feasibility statements are the central tool of the new ONDRAF/NIRAS strategy. Most of these statements are derived from the safety concept and are organised in a hierarchical tree structure. The top-level statements define the *a priori* objectives pursued by the programme stage in agreement with the boundary conditions. Lower-level statements setting out more detailed requirements of these objectives are derived from these top-level statements in a top-down approach. The substantiation of the safety and feasibility statements, with multiple lines of evidence and their associated uncertainties generated from the RD&D programme, is performed bottom-up. The assessment of the level of support given to the statements and the impact of the associated uncertainties to the higher-level statements provide an efficient and synoptic tool to structure the research and development activities and set the priorities between the issues to handle for a given decision. The multiple lines of converging arguments supporting every safety or feasibility statement show the robustness – in the sense of reliability – and the demonstrability of the claimed assertions.

The introduction of safety and feasibility statements in the Belgian programme resulted in significant changes. In particular, reorganisation of the RD&D programme around the need to substantiate the statements has led to a more efficient and pertinent structuring of the issues. The statements provide the means for a permanent consistency check between requirements on the system and its components, expected to become more stringent as the programme progresses.

The assessment of the feasibility of the EBS component as advised by the IRT was performed in this strategic framework. The reference design of SAFIR 2 was abandoned, mainly in view of its complexity in terms of feasibility and expected evolution. The development process of the new design has been the work of a task force co-ordinated by ONDRAF/NIRAS, involving people from different organisations and different fields of expertise. In the time period between October 2001 and December 2003, the working group elaborated three alternative basic designs (Supercontainer, Borehole and Sleeve) according to a step-by-step approach and with justification of the key decisions taken during the process. The selection procedure of the reference solution involved the performance of a multi-criteria analysis based on the safety principles, the boundary conditions and the strategic choices set in the strategy. The results gave a preference to the so-called “supercontainer” concept (ONDRAF/NIRAS, 2004). This concept was driven by the necessity to increase the longevity of the overpack. This strategic objective was reached by the choice of a cement-based backfill (termed the “buffer”) made of concrete based on Ordinary Portland Cement (OPC) surrounding the overpack and insuring a high-pH environment favouring passive uniform corrosion of the overpack. Apart from permanent shielding (which eliminates the need for remotely controlled underground operations and intrinsically better protects the workers), this concept also contributes largely to the system containment phase next to the clay host rock. Finally, it should be kept in mind that a concrete liner of the galleries is unavoidable due to the convergence of the Boom Clay host rock. The combination of high-pH concrete and carbon steel is also consistent with an ONDRAF/NIRAS strategic objective to use materials for which broad and proved experience and knowledge already exists, in order to reduce uncertainties linked to the long-term evolution of a material as much as possible (application of BAT principle). The safety and feasibility tree provides here an efficient tool to structure and identify inter-dependent links between different fields, in particular between long-term safety, operational safety and feasibility aspects to which the supercontainer must conform (ONDRAF/NIRAS, 2013b).

The IRT recommendations of SAFIR 2 to initiate dialogue with stakeholders were in line with the legal framework of 13 February 2006 relating to the assessment of the environmental impact of certain plans and programmes and public participation in their drafting. In accordance with this legal framework, ONDRAF/NIRAS drew up a “Waste Plan” and its associated Strategic Environmental Assessment (SEA) (ONDRAF/NIRAS, 2011). The Waste Plan compiled in a single document, all elements necessary to enable the government to make, with full knowledge of the facts, a decision in principle, that is a general policy decision or a general guidance decision, relating to the long-term management of B&C waste. In addition to public opinion, opinions on the Waste Plan were also requested from the SEA Advisory Committee, the Federal Council for Sustainable Development, the regional governments and the Federal Agency for Nuclear Control. Prior to the Waste Plan, ONDRAF/NIRAS elected to carry out a societal consultation broader than that required by the law. A series of dialogues and an interdisciplinary conference were indeed organised in 2009 in order to give civil society organisations, experts and interested citizens the opportunity to express their concerns and expectations with regard to the long-term management of B&C waste. For additional societal input, ONDRAF/NIRAS asked an independent body (King Baudouin Foundation) to organise a participatory process. Over three weekends a group of citizens debated the subject of the long-term management of high-level and long-lived radioactive waste.

The requested decision concerns the technical solution recommended by ONDRAF/NIRAS for the long-term management of B&C, namely a geological disposal in poorly indurated clay (Boom Clay or Ypresian Clays), in a single facility on Belgian territory and implemented as soon as possible, the pace of development and realisation of the solution being proportionate to its scientific and technical maturity, as well as to the public support it receives. Consideration of the stakeholder views collected as part of these participative processes resulted in ONDRAF/NIRAS matching its recommended management solution to conditions in terms of the waste's retrievability, monitoring the proper functioning of the disposal system and transferring knowledge about the facility, conditions for which the exact scope is still to be defined in future consultations with all interested parties. These stakeholder requirements are now included in the boundary conditions and translated within the safety and feasibility statements.

Following the waste plan, ONDRAF/NIRAS has decided to write a RD&D Plan on geological disposal (ONDRAF/NIRAS, 2013b), which presents the status of understanding and plans for the RD&D in preparation for the future safety cases. This plan also integrates societal concerns expressed in the frame of the consultation of the Waste Plan. This RD&D plan, to be published in the near future, proposes a structured R&D programme and is structured according to the safety and feasibility statements.

With hindsight, the NEA recommendations on SAFIR 2 reveal to have been very beneficial to the Belgian programme. They helped ONDRAF/NIRAS to optimise and structure its RD&D programme, but also to reorganise the way to perform research and development in a more safety-case-oriented way. The geological programme is now embedded in a societal dialogue with the regulatory body and other stakeholders. Governmental involvement, under the form of a decision of principle is now highly desirable to give a new trigger towards implementation.

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Implications of the regulatory framework and activities on R&D supporting repository implementation

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Introduction

In the context of geological disposal, the mission of the Belgian regulatory body is to ensure that the repository is developed, constructed, operated and closed in a safe manner, i.e. people and the environment are protected against the hazards of ionising radiation emitted by the radioactive waste, without imposing undue burdens on future generations.

This mission involves several types of activities such as the establishment of regulatory requirements as well as of procedures and conditions for meeting these requirements for the various stages of the licensing process (IAEA, 2011). The roles of the regulatory body also include the oversight of the activities of the organisation in charge of waste disposal and the review of the safety case and of its updates throughout the whole process of developing and implementing the geological disposal programme.

In order to fulfil its mission, the Belgian regulatory body (constituted by FANC and Bel V) carries out its own R&D programme. R&D work is essential for regulators as it allows maintaining and improving their scientific and technical skills, contributes to their independence and helps to build public confidence in the regulatory system.

The following sections give an overview of the Belgian pre-licensing and licensing process. The paper then discusses the possible implications of the regulatory framework and activities (R&D, safety case reviews and issuance of regulatory advices) on the R&D programme developed by the implementer of a disposal programme.

The pre-licensing and the licensing process

In 2011, ONDRAF/NIRAS submitted a Waste Plan to the Federal Government for a decision-in-principle, in order to set a policy for the long-term management of high-level and/or long-lived waste in Belgium. In July 2013, the decision of the government is still pending.

According to this Waste Plan the next milestone is the first Safety and Feasibility Case (SFC 1). The SFC 1 would be devoted to the assessment of the safety and feasibility of disposal systems that would be built in the Boom and Ypresian Clays and located in one or several potentially suitable zones with a view to supporting a decision of the type “go for siting”.

The second Safety and Feasibility Case (SFC 2) would ideally be site-specific and seek to provide evidence of the absence of any major safety- or feasibility-related obstacle to implementation. Based on the SFC 2, a go-ahead for launching the detailed site-specific studies needed to prepare the license application could be given.

The licensing process of a geological repository comprises different steps from the application for construction and operational phase up to the release of regulatory control. The first license describes the phasing and the conditions related to subsequent licenses. Each subsequent license confirms compliance with the conditions of the previous license and defines additional conditions for the next phase.

The regulatory framework

The Royal Decree of 20 July 2001 (MB, 2011a) (laying down the general regulation on the protection of the population, the workers and the environment against the hazards of ionising radiation) outlines the main regulatory provisions that operators of nuclear facilities must comply with. According to this Royal Decree, radioactive waste disposal facilities are Class 1 facilities.

The Royal Decree of 30 December 2011 (MB, 2011b) (laying down the safety requirements for nuclear installations) outlines the general safety requirements for all Class 1 facilities.

However, the existing regulation does not cover the aspects specific to disposal facilities, e.g. the long-term aspects of radioactive waste management and the stepwise approach required to manage the licensing process over large time scales are not addressed. Therefore, Royal Decrees specifying the licensing system and the specific safety requirements for disposal facilities were developed by the FANC and are now in the stage of approval and promulgation.

FANC has also developed technical guides specific to radioactive waste disposal. These documents provide guidance on the interpretation and implementation of safety and radiation protection principles and requirements specified in the regulation. They include recommendations and regulatory expectations on the development of the safety strategy, the development and implementation of a disposal facility and the safety assessment.

Safety and radiation protection principles

The Belgian regulatory framework for radioactive waste disposal is underlain by two safety principles (the defence-in-depth and the demonstrability principles) and the radiation protection principles of the ICRP (ICRP, 2007; Weiss, 2013).

The application of the principle of demonstrability has direct implications for R&D as this principle implies that the implementer of a disposal programme:

- demonstrates that the disposal facility can be constructed with the required level of performance (i.e. feasibility of its construction);
- uses proven techniques or new techniques based on qualification tests;
- demonstrates that the effective performance of the disposal system (i.e. as-built performance) allows to protect people and the environment against the hazards of ionising radiation despite all perturbations which might reasonably be expected and construction contingencies;
- demonstrates that uncertainties are correctly managed.

It is clear that the outcomes of the R&D programme carried out by the implementer constitute an essential pillar of the demonstration of the performance and feasibility of the disposal system. Consequently, the implementer should permanently consider the various implications of the principle of demonstrability when developing its R&D programme.

The ICRP principle of optimisation of protection when applied to the development and implementation of a geological disposal system has to be understood in the broadest sense as an iterative, systematic and transparent evaluation of options for enhancing the protective capabilities of the system and for reducing radiological impacts (ICRP, 2007; Weiss, 2013).

The implementer's R&D programme provides essential data and information to identify, assess and compare options. The development of this programme is therefore strongly linked to the continuous process of optimisation of protection.

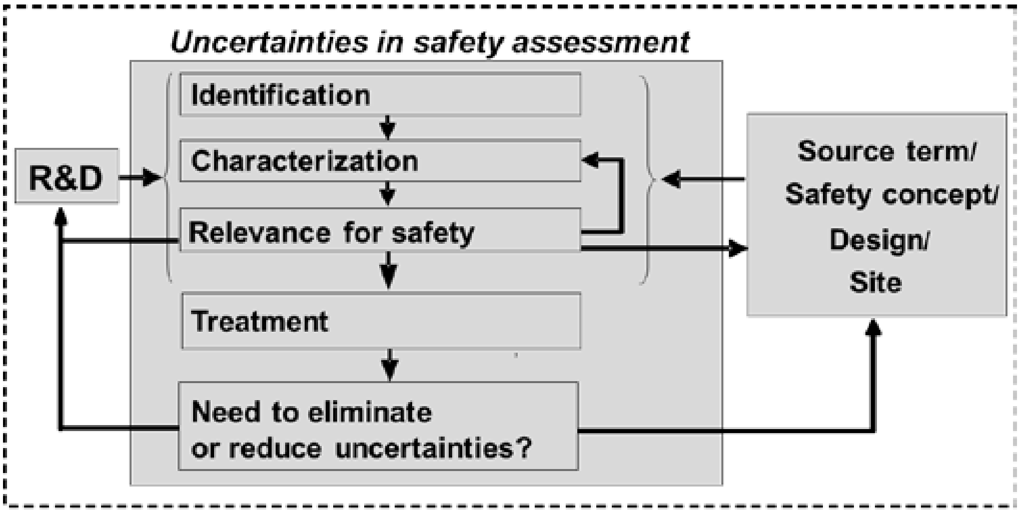
Safety requirements

The regulatory body has developed a set of requirements that the implementer has to fulfil in order to develop, operate and safely close a disposal system. Some of these requirements can have direct repercussions on its R&D programme. Indeed, the outcomes of this programme are an essential input for the argumentation that these regulatory requirements are met.

The development and implementation of a safety strategy is a key safety requirement. The safety strategy is intended to define the objectives and principles guiding the overall disposal programme.

The safety strategy addresses a number of key elements such as optimisation of protection, defence-in-depth through the provision of multiple safety functions and of robust repository components, the containment and isolation of the waste, the use of passive safety features, and the demonstrability of safety-related features. It should also define the approach that will be followed to assess safety and manage uncertainties. Figure 1 illustrates the role of R&D in the management of uncertainties (FANC, 2012). It shows that R&D is necessary for the identification and characterisation of uncertainties as well as for the understanding of their safety relevance. R&D may also be needed to reduce uncertainties the magnitude of which magnitude do not allow demonstrating the safety of the disposal system.

Figure 1: Management of uncertainties



As such, the safety strategy is the starting point to develop a geological repository and therefore to define and conduct the R&D programme. More specifically, the safety strategy should identify the objectives of the R&D programme and explain how this programme is integrated in the repository development process so as to ensure that these objectives will be reached.

Regulatory requirements specific to the engineered barriers, the host rock and the site are also important elements to be considered by the implementer when developing its R&D programme.

Additionally, requirements related to safety assessment addressing the following topics have a significant influence on this programme:

- building confidence in the assessment;
- performance assessment (i.e. the ability of the system and of its components to fulfil their safety functions);
- radiological impact assessment.

The necessity to establish confidence in the assessment derives directly from the demonstrability principle. This implies among others that (FANC, 2012):

- the assessment rests on best available knowledge;
- the disposal system is well understood;
- the identification and treatment of FEP is traceable and well-founded;
- a set of scenarios representative and bounding of the possible evolutions of the system is developed;
- models are shown to be appropriate to the objectives of the modelling through a justification, verification and validation process;
- uncertainties are properly identified, characterised, analysed, treated and assessed as illustrated in Figure 1.

To fulfil these requirements, the implementer has to develop its R&D programme with the objective of establishing a sound assessment basis. This implies that the R&D programme leads to a:

- proper identification and understanding of the safety-relevant phenomenology (i.e. processes upon which safety functions rely, events and processes that may affect safety functions and radionuclide transport);
- reliable characterisation of the disposal system and its environment;
- appropriate model verification and validation;
- reliable identification, characterisation and analysis of uncertainties.

Finally, the regulatory body requires that the implementer develops and implements a quality assurance programme. This also applies to the R&D programme and more specifically to the performed tests and experiments. In particular, the verification of the reproducibility of the measurement and experimental results is an important issue.

R&D programme in support of regulatory activities

According to the Law of 14 April 1994 on the Protection of the Public and the Environment Against Radiation (FANC, 1994), FANC is responsible for maintaining scientific and technical documentation in the area of nuclear safety and radiological protection. It is also responsible for fostering and co-ordinating R&D and establishing relationships with national and international research organisations.

The R&D objectives set by the regulatory body differ generally from those set by the implementer. The regulatory body will mostly investigate issues directly related to safety with the objective to verify the adequacy of the approaches followed by the implementer to reach the safety objective. The regulatory body may decide to initiate R&D work where it considers that there is a need for additional studies beyond those undertaken by the implementer. There may also be situations in which the regulatory body needs independent R&D work in order to perform a critical and objective review and assessment. Special attention in R&D programmes will usually be given to the detection of possible inadequate choices, assumptions, knowledge gaps, incompleteness, inconsistencies, mistakes (of reasoning or of implementation),…. R&D activities performed by the regulatory body also help to increase the credibility of its technical competence, integrity and judgement.

These activities are therefore more a "complement to" and a "verification of" than a "duplication of" the R&D activities performed by the implementer. Nonetheless, in certain situations they can interfere with each other, especially when a regulatory body's findings lead to different conclusions than those put forward by the implementer.

Safety case reviews

The regulatory body has a continuing role to review the safety case which has to be regularly updated to remain an adequate basis for making decisions throughout the repository life cycle.

The review aims to determine whether the safety case has been developed to an acceptable level in terms of quality and confidence in safety to move to the next phase of the project.

This includes the verification that the safety case complies with the "regulatory framework". More specifically, the regulatory body will evaluate whether the safety case provides an adequate and appropriate basis to demonstrate that the proposed facility will be operated safely and provides reasonable assurance of an adequate level of safety in the period after closure. So, the regulatory body has to verify, among other things, that the implementer's argumentation and assessment basis rest on the findings of a sound R&D programme.

Another specific objective is to evaluate if the proposed R&D programme contributes adequately to the management of uncertainties. More specifically, the regulatory body verifies that relevant measures for mitigating uncertainties have been identified and addressed, and that adequate follow-up plans for their implementation have been put into place.

Hence, the recommendations of the regulatory body resulting from the review of a safety case will generally have a significant impact on subsequent phases of the R&D programme conducted by the implementer.

Regulatory advice

To ensure adequate and efficient steering of repository development, regular interactions between the regulatory body and the implementer on specific safety-related issues are generally needed. Such interactions can lead to formal recommendations and allow sharing views on:

- the interpretation of existing international recommendations;
- the methodological approaches to assess operational and post-closure safety;
- the scientific and technological bases needed to move to the next step of the repository development programme.

It is thus obvious that these interactions can also have repercussions on the R&D programme of the implementer.

Conclusions

Regulatory bodies are responsible for the establishment of a regulatory framework specifying the requirements and conditions for the development, operation and closure of disposal facilities. Performing an independent verification of compliance with these requirements and conditions involves different types of activities such as reviews, issuance of advice and R&D activities. These activities together with the regulatory framework can lead to recommendations addressing directly or indirectly the R&D programme of the implementer.

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Session 3.4

Safety Case for License Application for a Final Repository: The French Example

Safety case for license application for a final repository: The French example

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The reversible repository in a deep geological formation is the French reference solution for the long-term management of high-level and intermediate-level long-lived radioactive waste (HLW and ILW). Twenty years of R&D work and conceptual and basic studies since the first French Act of 1991 led, in particular, to a feasibility demonstration in 2005. According to the French Act on Radioactive Waste of 28 of June 2006, Andra shall design a reversible repository in order to apply for license in 2015. In response to this demand, Andra developed the industrial project known as "Cigeo", a reversible geological disposal facility for HLW and ILW located in Meuse/Haute-Marne.

Two years before applying for authorisation, Andra's project is now focusing on three main targets: developing Cigeo's industrial design, preparing the authorisation process through increased exchanges with stakeholders and the preparation of a safety case to support authorisation application. The latter draws on the previous safety cases of 2005 and 2009, which give a sound basis to assess Cigeo's safety, both for the operational and post-closure periods. In this new stage of the project, the challenging issues for the preparation of the safety case are the following:

- to identify the various regulatory frameworks (nuclear and non-nuclear) and guides applicable to the facility;
- to ensure that the industrial design complies in particular with the safety requirements as presented in the safety case and its supporting safety assessment;
- to identify crucial inputs (R&D, tests,...) needed to support the authorisation application, in particular, to bring convincing arguments to assess the technical feasibility of the design and when appropriate its ability to meet the safety requirements;
- to ensure that all the requirements from previous regulatory and peer reviews (national and international?) are taken into account.

General schedule

Thanks to the first outputs of the industrial design, Andra was in a position to propose to the Public Debate National Commission, an independent body tasked with organising public debate on large infrastructure projects, to hold the public debate on Cigeo in 2013. The debate began in May and will end in December 2013.

Any changes made to the project following the public debate, along with the avenues for optimisation identified by Andra will be taken into account in the following study phase before the repository license application is filed in 2015.

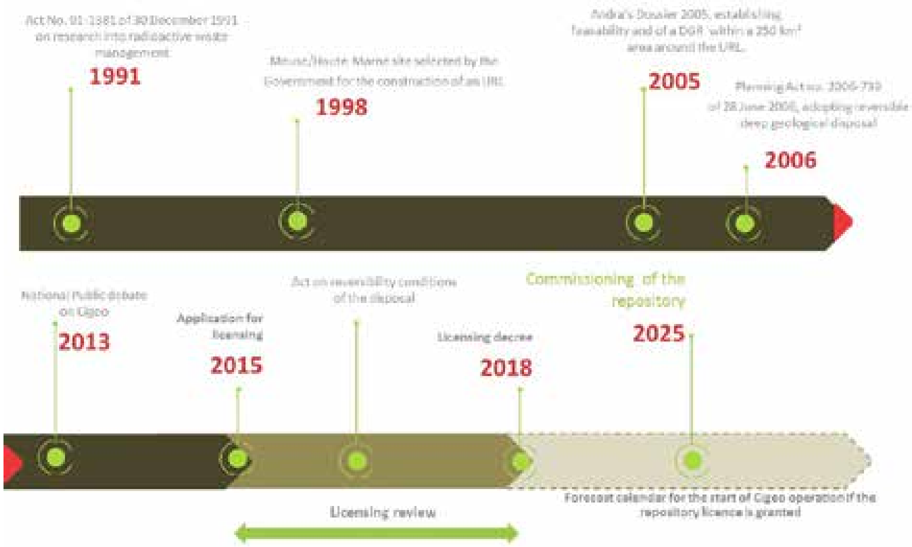
The content of the application file is defined by Decree (2 November 2007). It comprises various plans of the facility, the description of the solution envisaged for the closure of the facility, the preliminary safety case (operational safety, long-term safety, protection against malicious acts, management of accidents, preliminary acceptance criteria,…), the environmental impact assessment (health, transport, human activities, nature, patrimonial aspects,…). These elements shall be produced on the basis of a sufficiently detailed design, so that the safety authority can fully grasp their industrial feasibility and their required performance.

The regulatory review process consists of many evaluations and decisions: The license application is examined by the National Assessment Board, Nuclear Safety Authority; local authorities are consulted. An assessment is carried out by the Parliamentary Office for the Evaluation of Scientific and Technological Choices. An Act shall be passed on the disposal reversibility conditions. Afterwards, Andra shall update its license application to take into account this new law, before the review process is achieved: examination by the Nuclear Safety Authority and its technical safety organisation (TSO), the Institute of Radioprotection and Nuclear Safety (IRSN) and public inquiry will be carried out prior to the “Conseil d’Etat” decree granting the repository license for Cigeo.

According to French regulation, a creation decree will authorise the construction of the facility and then the nuclear operations to be performed. An explicit and unequivocal safety demonstration has to be provided for these operations in the license application. In the case of Cigeo, the authorisation for operations foreseen in the far future may be granted on the prerequisite that complementary dedicated files be transmitted in the future. This regulatory mechanism may be generalised to all operations for which complementary elements are found necessary by the regulator.

According to the planning Act of 26 June 2006, the commissioning of the facility is planned by 2025. The nuclear operations will start with the reception of the first waste package provided the commissioning of the repository is authorised by the safety authority. At this stage, the commissioning encompasses only the first portion of the facility. Beyond 2025, construction and equipment work will be carried out concurrently with nuclear operations in the previously commissioned portions.

Figure 1: General schedule for French deep geological repository Cigeo



Cigeo's general layout

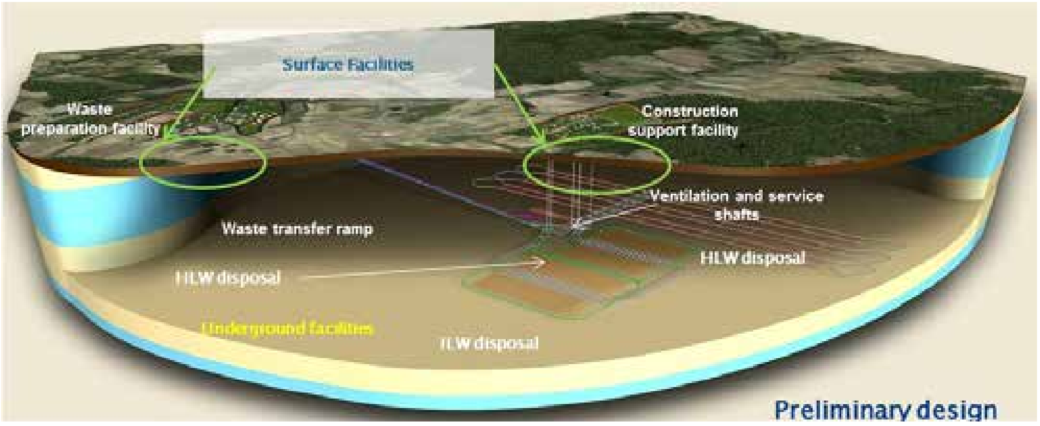
Cigeo will consist of surface installations for operations such as waste package receipt, inspection and preparation, an underground disposal installation, and an infrastructure that will connect the underground installation with the surface. The repository will operate for more than 100 years and be built according to a step-by-step planning process over time. To guarantee its role and ensure that waste is confined over very long time periods without the need for human intervention, the underground structures at Cigeo need to be closed up. This closure will take place gradually.

Located at a depth of around 500 meters, the Cigeo underground facility will be expanded over the course of its operations. It will consist of separate disposal zones for HLW and ILW-LL, connecting drifts and technical facilities. By its 100th year of operation, Cigeo's footprint will be around 15 km². Robots will emplace waste packages in horizontal tunnels, known as cells, excavated in the heart of the Callovo-Oxfordian layer. HLW will be emplaced in metal-lined cells measuring a few hundred meters in length and around 70 cm in diameter. ILW-LL will be emplaced in horizontal disposal cells measuring a few hundred meters in length and around ten meters in diameter. The disposal zones will be modular in design to allow waste disposal tunnels to be built according to a step-by-step planning process over time.

Two types of infrastructure will connect Cigeo's surface installations with the underground facility. Vertical shafts will be used to transfer workers, construction equipment and materials, and ventilate the underground facility. Waste packages will be transferred by means of a funicular along an access ramp.

Cigeo's surface installations will be split across two sites (ramp zone and shaft zone) located a few kilometres apart. The ramp facility will be built in the zone already prepared around the current URL and integrated with the site's physical features.

Figure 2: Cigeo's general layout after 100 years of operation



Upon arriving at Cigeo, waste packages will be transferred to buildings where they will be taken out of their transport casks and inspected (absence of contamination, dose rate, etc.). These buildings will also be used to manage flows of waste packages prior to their transfer into the underground facility. These facilities do not intend to replace waste producers' own storage facilities, particularly those playing the role of cooling HLW sufficiently prior to transfer and emplacement into the repository. Waste packages will then be placed in disposal containers. Disposal packages will be placed in a cask to shield against radiation. The cask will be loaded onto a funicular that will slowly descend all the way to the galleries. The cask will then be transferred to the cells and the emplacement of waste packages in the cells may be remotely controlled. The transfer cask will dock with the interface door of the cell in order to create a containment system while the door is open. The door will not be able to be opened until the transfer cask is correctly docked. The handling system will then transfer the waste packages into the cell and the door of the cell will be closed in order to protect workers from exposure and to fulfil containment requirements.

Safety case in the industrial design phase and preparing for licensing

The 15-year period from 1991 to 2006 was mainly dedicated to research activities, especially thanks to the underground repository laboratory (URL) constructed at the border between the Meuse and the Haute-Marne districts, at the Bure location. Following the 2006 Planning Act, work progressively focussed on the siting process for Cigeo and on preparing for the project's industrial development.

In late 2009, Andra submitted the so-called "Dossier 2009" to the French government. The Dossier included proposals concerning the siting and design of Cigeo, as well as an intermediate preliminary safety case focused on operational safety.

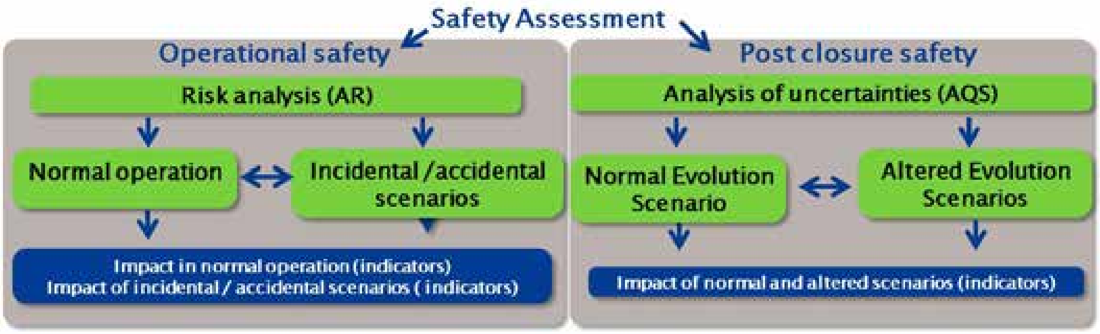
After the review of this file and decisions taken by the government on the siting, Andra decided to launch the industrial design phase in 2011. For this new stage of the project, Andra set up a dedicated industrial structure for Cigeo project development. To reinforce the engineering resources of the project team, Andra engaged a prime contractor, the Gaiya group formed by Technip and Ingerop. An industrial outline was developed in in 2012. A detailed design will be established from 2013 on. In its position of implementer and future operator, Andra shall ensure a robust technical control of the project throughout its existence. Andra has thus reinforced its engineering skills (project management, design and handling of disposal packages, infrastructure engineering, economic assessment), by creating a specific division for engineering and managing the Cigeo project, with a view to ensuring the operational follow-up of subcontracts on studies.

This new organisation imposes specific measures to ensure that safety requirements established on the basis of previous safety cases are still met during the development of the industrial design. To achieve that target, Andra has established and characterised functions to be fulfilled by the components of Cigeo and also frozen design choices, in order to leave space for optimisation of the industrial design but at the same time to preserve critical safety parameters. Short-term and operational requirements are expressed mainly on a safety functional basis (for example: to meet workers' radiological protection targets against potential exposure); on the contrary, technical solutions to long-term issues, which are out of scope of classical nuclear engineering, are in some cases imposed and specified along with functional requirements. These technical solutions to fulfil post-closure safety functions were established thanks to Andra's 20 years' experience in long-term radioactive waste management. For example, the location and length of seals are imposed, but the choice of the material is left open, provided an overall performance is achieved (permeability, swelling pressure).

To establish those requirements, Andra also had to analyse applicable regulatory frameworks, guides and practices, in order to identify potentially conflicting regulations and practices (e.g. nuclear versus mining) or a lack of references. For example, specific requirements on fire risk were identified in order to take into account the combined constraints of "conventional" underground facilities (tunnels, mines) and nuclear facilities. Andra was thus obliged to establish specific guidelines on handling fire risk for the underground nuclear facility, as no existing guidance was directly applicable. Other specific requirements during the operational period have been identified and concern, for instance, enabling proper management of the containment systems or the co-activity (co-existence of conventional and nuclear activities in the underground facility).

On the basis of the developed industrial design, Andra is preparing the safety case in support of the license application. As previously said, the safety case shall be produced on the basis of a sufficiently detailed design, so that the safety authority can fully grasp its industrial feasibility and its compliance with the required performances. To do so, Andra draws on the iterative process employed up to now, which provides a sound methodological basis. Concerning operational safety, the Cigeo safety case will follow standard practices for risk management for classical nuclear facilities in accordance with French regulations. Therefore, Andra will apply a classic risk analysis scheme in its assessment of normal, incidental and accidental scenarios. Concerning post-closure safety Andra is considering the development of a Qualitative Safety Analysis which would define normal and altered evolution scenarios. This methodology would be strengthened through the integration of preceding reviews' outcomes and of recent developments in international standards, guidance and best practices. For example, to improve the comprehensiveness of the safety case, the synergy between FEP and functional analysis will be deployed at the outset of QSA.

Figure 3: Structure of the safety assessment



An operational safety case was outlined in the "Dossier 2005" and more thoroughly assessed in the "Dossier 2009". Central issues such as the definition of containment systems, fire risk management and co-activity management were identified, so that Andra was able to designate relevant requirements pertaining to these issues as input for the industrial design. Cigeo's ability to meet legal performance requirements will be assessed on this basis.

Regarding the post-closure phase, it was thoroughly developed in the "Dossier 2005". The main tasks for the safety case supporting the license application are manifold:

First, a formal process must define what the inputs for the safety case are: what is the state of knowledge, what is the waste inventory considered for safety assessment, what is the reference design? This task is all the more important so that during the industrial design development phase, the layout of the facility may be optimised and design options may be closed or left open. The impact of such evolutions on the safety must be controlled. To do so, formal intermediate reviews in line with the planning of the industrial design development are organised throughout the preparation of the safety case.

On this basis, Andra must establish the evolution of knowledge and the remaining uncertainties, after ten years of supplementary work in R&D and experiences in the Bure URL since the publishing of the "Dossier 2005". Andra must also consider the evolution in design since 2005, and thus reassess whether remaining uncertainties are managed by technical components or by scenarios.

Finally, Andra must bring convincing arguments to assess the technical feasibility of the design and its ability to meet the safety requirements. To do so, R&D results have been completed by a large set of technological tests and developments of various topics (e.g. mining, mechanical, waste containers, sealing systems) in the underground laboratory in Bure as well as in surface laboratories in order to verify parameters, develop equipment, optimise disposal solutions and tackle potential emerging issues. A programme of technological tests and demonstrations, identifying the results needed for the license application, has been developed, and further tests are planned over the long term to confirm, if needed, performance of components with full-size tests in the facility before its commissioning. From the outcomes of those tests, Andra also aims to provide a solid and documented technical outcome as a back-up to the Cigeo licensing application, responding to themes and issues identified by evaluators in previous reviews. In particular, given the importance of seals for the post-closure safety of the facility, a number of tests have been implemented to demonstrate their industrial feasibility and performance.

How a regulator is preparing for reviewing a license application file: The case of ASN

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French context

ASN

The French Nuclear Safety Authority (ASN) is an independent administrative authority. It prepares regulation pertaining to the management of radioactive waste, monitors the control of safety of basic nuclear installations that produce or treat waste or are involved in their disposal and performs inspections of waste producers (EDF, AREVA, CEA, hospitals, research centres, etc.) and Andra, the French National Radioactive Waste Management Agency. It regulates the overall system set up by Andra for accepting waste from producers and assesses waste management policy and the practices of radioactive waste producers. It reviews license applications and authorises commissioning of nuclear installations.

In order to review technical documents, ASN benefits from the expertise of technical support organisations. The French Institute for Radiation Protection and Nuclear Safety (IRSN) is the main such organisation. ASN has been making efforts to diversify its experts for several years.

In preparing its decisions, ASN also calls on the opinions and recommendations of seven Advisory Committees of Experts (GPE), with expert knowledge in the areas of waste, nuclear pressure equipment, medical exposure, non-medical radiation protection, reactors, transport, and laboratories and nuclear plants. ASN consults the GPEs in preparing its main decisions. In particular, they review the preliminary, provisional and final safety analysis reports for each nuclear installation. They can also be consulted about changes in regulations or doctrine.

Regulatory framework with regard to deep geological disposal

Researches and studies related to the deep geological repository are performed in the framework of French law.

Firstly, the Act of 30 December 1991 related to research on management of radioactive waste defined three different axes of research on management of high-level radioactive waste: separation and transmutation, deep geological repository (both reversible and irreversible) and long-term storage. Andra was in charge of research on deep geological repository.

In order to perform this research, Andra was authorised, in 1999, to operate an underground laboratory at the border between the Meuse *département* and Haute-Marne *département* in Bure.

Some milestones were defined and Andra submitted files at each step with a presentation of the work carried out.

In 2005, Andra submitted the "Dossier Clay-2005" that was reviewed by ASN with the support of its technical support, IRSN and of the advisory committee in charge of waste. ASN issued a stance on this work on 1 February 2006.

The Act of 28 June 2006 on sustainable management of radioactive materials and waste defined a reversible deep geological repository as the reference solution for management of radioactive waste that cannot be disposed of in surface or near-surface repositories. Andra is in charge of designing, sitting, operating, decommissioning and monitoring this deep geological repository. The law defines a schedule for the commissioning of this disposal facility: an application file for creation authorisation must be reviewed in 2015 and according to the result of the authorisation process, this disposal facility should be commissioned in 2025.

This "Waste Act" also determined that a national plan for radioactive materials and waste management (called PNGMDR) shall be produced and updated every three years. The purpose of the PNGMDR is to review the existing management procedures for radioactive materials and waste, to identify the foreseeable needs for storage and disposal facilities, to clarify the necessary capacity of these facilities and the storage durations and, for radioactive waste for which there is as yet no final management solution, to determine the objectives to be met. The main provisions of the plan and the studies required by the PNGMDR are set by a ministerial decree.

This plan is co-directed by ASN and the Ministry of Energy.

This plan is used as a steering document for the different steps of designing and sitting a deep geological repository.

Safety guide

In 2008, ASN issued a safety guide related to disposal of radioactive waste in a deep geological repository.

This guide defines objectives that have to be met from the early investigation phase for siting and conception of a repository in order to ensure long-term safety after closure of the installations.

This guide is a way to frame works performed by Andra.

The guide, resulting from a working group of experts organised by ASN, was issued after comments of the advisory committee on waste management.

How ASN prepares for reviewing the license application

Staff

ASN does not have a full team dedicated to the deep geological repository, but has recruited a project manager to create a link between all the subjects and entities involved in the project (regulatory, fire specialists, communication, etc.). This project manager dedicates part of his/her time to participate in international working groups or conferences in order to enlarge his/her view of the subject.

Moreover, ASN has to maintain and improve the balance of expertise and experience of its advisory committees, particularly the one dedicated to waste that will be the most implicated in the review of the license application.

With this objective, ASN organises with the support of IRSN dedicated meetings for its advisory committee in order to provide it with generic technical or scientific information related to deep geological repository and to present the latest work from Andra that is not directly submitted to its recommendation (only main decisions require advice from advisory committees).

These meetings enable it to follow Andra's work and to receive information it should require during document review. The advisory committee can also directly exchange with international peers (e.g. yearly meetings with German peers) or by presentation of work carried out in international projects (e.g. presentation last spring of works performed in the PRISM project).

IRSN also recruits and trains its own staff in order to prepare itself for reviewing the application file. Moreover, IRSN carried out research in order to check data presented by Andra and to improve the competence and experience of its own staff.

Reviewing early files

Within the framework of PNGMDR, Andra and waste producers can be asked to carry out studies and issue associated technical documents. Several studies are thus requested every three years (period covered by a version of the PNGMDR).Then, for each of them, the stance of the ASN can be required by the Ministry of Energy.

In addition, Andra can also submit documents to ASN, for instance defining options it is considering for the design, site, operation and monitoring of a disposal facility. ASN can decide to issue an opinion on these documents.

Since 2006, ASN issued five main opinions on documents issued by Andra:

- In 2006, on the results of studies on the three axes of research defined by the "Waste Act" of 1991 (separation and transmutation, long-term storage, deep geological repository). In this opinion, the ASN defined deep geological repository as an essential solution for management of high level waste.
- In 2010, on a proposition from Andra on a specific area for further research (ZIRA) in view of siting of a deep geological repository (as requested by the PNGMDR).
- In 2011 on the dossier "Jalon 2009" issued by Andra on extension of operation in the Bure underground research laboratory.
- In 2013 on a proposition of inventory, seismic investigation results and research related to disposal of spent fuel and answers from Andra on a study performed by an independent organism (IEER) by request of a local committee implicated in Andra's work.

Each of these opinions was an opportunity for ASN to:

- evaluate the quality of the work carried out by Andra;
- request further studies on specific points;
- define elements that have to be developed for the license application file;
- define principles according to which license application will be reviewed.

For instance, in its latest opinion ASN required Andra to consider a specific scenario in its safety case, asked Andra to generalise the use of both determinist and probabilistic approaches and provided Andra with principles that have to be followed in order to establish the waste inventory in view of the license application file.

Regulatory aspects

A deep geological repository is a specific nuclear installation, given the fact that part of it is underground and that it is intended to be operated for more than 100 years.

Moreover, the principle of reversibility and retrievability introduces new needs during application file reviewing.

In consequence, the licensing of such a facility will require a specific procedure.

In France, the “Waste Act” of 2006 introduces news steps in the authorisation procedures: a public debate (which was held between March and December 2013), specific consultation of local communities, the need for a law defining the conditions of reversibility of the facility – a law that must have been voted by the French parliament before authorisation could be granted,...

The implementation of these principles must be explained in a ministerial decree. ASN and the Ministry of Environment are currently working on that document.

In addition, exchanges between ASN and Andra are required, so that ASN can define the content required for the application file.

However, regulatory aspects are not limited to the authorisation process of the facility. The whole lifetime of the installation requires specific arrangements: modification of authorisation (e.g. in case of evolution of the inventory for instance to allow disposal of spent fuel), follow-up of reversibility, backfilling, sealing or closure of parts of the repository, final closure (in France this would require a specific law) and oversight.

All these aspects must be studied before the authorisation so as to share information with the public, and also due to their impact on the content of the application file.

Andra R&D programme

In 2011, Andra was authorised to pursue operations of its underground research laboratory.

This authorisation was approved after an opinion was issued by ASN.

ASN considered that the programme of R&D submitted by Andra was likely to enable it to gather the input data requested for filing its application file.

During regular inspections, ASN checks that the research carried out by Andra is consistent with its programme of R&D and is performed under good conditions.

International works

In order to prepare for the review of Andra’s license application file, ASN also relies on international works in different ways:

- to define principles (for instance on reversibility);
- to define a regulatory framework (for instance within WENRA);
- to improve its technical and scientific background;
- to exchange information with other regulators (in bilateral or multilateral meetings)

Session 3.5

Safety Case for License Application for a Final Repository: The Swedish Case

SKB's safety case for a final repository license application

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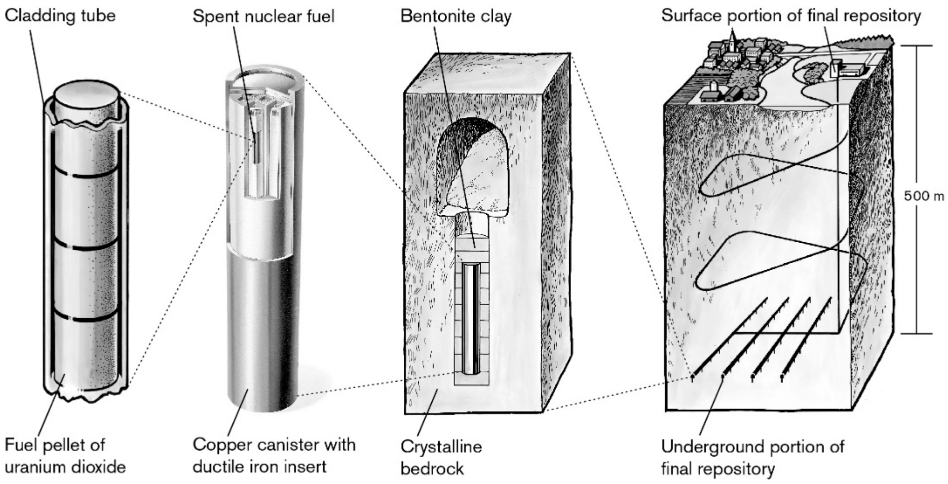
Background and context

The safety assessment SR-Site (SKB, 2011) is a main component in SKB's license application, submitted in March 2011, to construct and operate a final repository for spent nuclear fuel at Forsmark in the municipality of Östhammar, Sweden. Its role in the application is to demonstrate long-term safety for a repository at Forsmark. 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).

Figure 1: The KBS-3 concept for final storage of spent nuclear fuel



The principal regulatory acceptance criterion, issued by the Swedish Radiation Safety Authority (SSM), requires that the annual risk of harmful effects after closure not exceed 10^{-6} for a representative individual in the group exposed to the greatest risk (SSM, 2008b). SSM's regulations also imply that the assessment time for a repository of this type is one million years after closure (SSM, 2008a, 2008b). The licence applied for is one in a stepwise series of permits, each requiring a safety report. The next step concerns a permit to start excavation of the repository and requires a preliminary safety assessment report (PSAR) covering both operational and post-closure safety. Later steps include permission to commence trial operation, to commence regular operation and to close the final repository.

The assessment methodology

SKB's licence application is structured to meet Swedish regulations, and long-term safety is reported in detail in the SR-Site main report (SKB, 2011). Most elements of a safety case according to the NEA definition (OECD/NEA, 2004) appear in the SR-Site report hierarchy, whereas a few remaining elements are found elsewhere in the licence application. The assessment methodology applied in SR-Site consists of 11 main steps. The structure and content of the safety assessment has been developed in the context of the international progress in the field (OECD/NEA, 2009). The SR-Site assessment methodology is also influenced by the fact that the primary safety function of the KBS-3 concept is containment throughout the one million year assessment period. Key features of the assessment methodology include:

- The establishment of a quality assured initial state of the engineered components of the repository and of the site. The former is reported in several so-called "production" reports covering the spent fuel, the canister, the buffer, the deposition tunnel backfill, the repository closure and the underground openings constructions, respectively (SKB, 2010e, 2010f, 2010g, 2010h, 2010i, 2010p). A key step in the establishing of the initial state of the engineered components is the development of a set of design premises. Feedback from the preceding safety assessment SR-Can (SKB, 2006) was developed into design premises for the SR-Site assessment and the license application (SKB, 2009). Examples of important aspects of the initial state of the engineered barriers include: *i)* the copper canister sealing quality; *ii)* the cast iron insert casting quality; *iii)* buffer properties such as density and content of montmorillonite and impurities; *iv)* backfill properties ensuring its ability to keep the buffer in place and to swell; *v)* the quality of the approach to adapt the repository to the detailed conditions found underground and the quality of the excavation technique; *vi)* the quality of the deposition technique.

  The initial state of the site is reported in a descriptive model of the Forsmark site (SKB, 2008), that describes the results of the surface-based site investigation and the site modelling based on the site investigation data. The main safety-related features of the Forsmark site are: *i)* a low frequency of water conducting fractures at repository depth; *ii)* favourable chemical conditions, in particular reducing conditions at repository depth and salinity that would ensure stability of the bentonite clay buffer; *iii)* the absence of potential for metallic and industrial mineral deposits within the candidate area at Forsmark. In addition, the relatively high thermal conductivity at the site facilitates an efficient use of the rock volume and the rock mechanics and other properties of importance for a safe and efficient construction of the repository are also favourable.

- The compilation of the understanding of processes relevant for long-term safety for the spent fuel and the canister, buffer and backfill, geosphere and external conditions (SKB, 2010b, 2010c, 2010k, 2010l).

- The establishment of safety functions for the repository system, through which it is clarified which roles a particular component, e.g. the canister, has for safety. When possible, quantitative criteria for the fulfilment of safety functions are also provided.

- The detailed analysis of a reference evolution, yielding an understanding of the development of the repository components and of the system as a whole. The containment potential of the canisters is in focus whereas radionuclide transport and dose, should canister failures occur, are analysed in subsequent steps. The analysis of the reference evolution includes comprehensive modelling of thermal, hydraulic, mechanical and chemical processes in the repository and the surrounding host rock. The reference evolution is analysed in four time frames; *i)* the excavation/ operational period; *ii)* the initial c. 10 000 years' period of temperate domain of a

120 000 year reference glacial cycle; *iii)* the remaining part of the glacial cycle; *iv)* subsequent assumed repetitions of the reference glacial cycles up to one million years after repository closure.

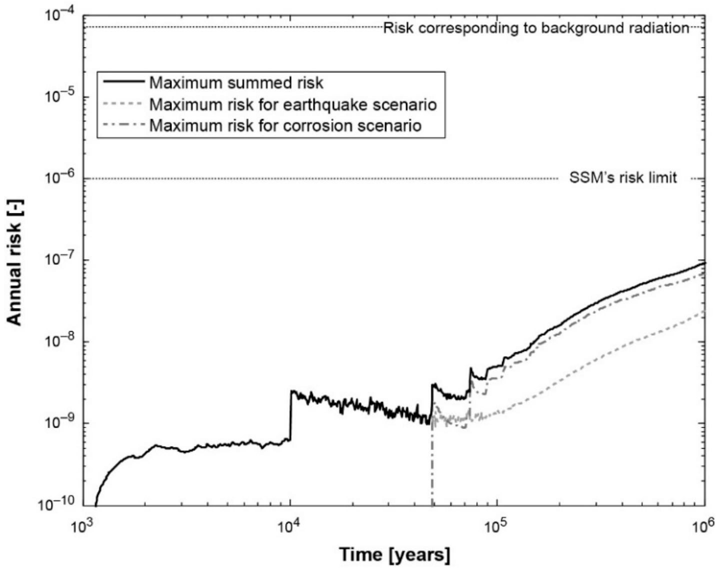
- The selection and analysis of a number of assessment scenarios. The selection is based on the safety functions and on the understanding of the repository evolution. The scenarios aim at examining all possible routes to canister failure. Scenarios for which canister failures cannot be ruled out, and for which releases of radionuclides thus occur, are propagated to probabilistic consequence calculations (SKB, 2010a, 2010o, 2011). The results of these calculations determine the fulfilment of the regulatory risk criterion. Future human actions are analysed as a set of stylised scenarios (SKB, 2010m).

Key results

The analyses in SR-Site (SKB, 2011) indicate that containment is maintained even as long as the one million year perspective for a vast majority of canisters. Deterioration of the barrier system to the extent that containment is lost is assessed to only occur, as a statistical average, for less than one canister out of the 6 000 for the scenario that yields the highest risk contribution. The two scenarios that yield contributions to the calculated risk in SR-Site are described below.

In a one million year time perspective, there is a small risk contribution from canister failures due to enhanced corrosion following buffer erosion. Loss of buffer may occur from exposure to low ionic strength waters but the extent is uncertain. The Forsmark site has a large potential to maintain a sufficient ionic strength at repository depth over a glacial cycle. Loss of buffer mass, to the extent that advective conditions arise in the deposition hole, may, however, occur in a 100 000 year perspective for typically less than ten deposition positions with high flow rates. Advective conditions in a deposition hole will enhance the canister corrosion rate. In a one million year time perspective, this may lead to failures of a few canisters when applying the most pessimistic of the hydraulic interpretations made of the Forsmark site. There is also a small risk contribution from a scenario where earthquake induced secondary shear movements in fractures intersecting deposition holes cause mechanical canister failures. This failure probability is lower than that for the corrosion scenario, even with a number of pessimistic assumptions.

Figure 2: Risk curves for the corrosion and earthquake scenarios, and their sum



Deterministic and probabilistic radionuclide transport and dose calculations were undertaken for a number of cases exploring uncertainties of the two risk contributing scenarios, and additional “What if?” scenarios (SKB, 2010o). The bounding, dashed curve in Figure 2 is the sum of the pessimistically calculated risks associated with the earthquake and corrosion scenarios. Since this curve is below the risk limit for the duration of the one million year assessment period, the analysed KBS-3 repository at the Forsmark site is assessed to fulfil the regulatory risk criterion.

Confidence in the achieved results

The confidence in the results obtained is in the SR-Site safety report (SKB, 2011) assessed as sufficient for the decision at hand based on the following:

- Knowledge of the Forsmark site from the completed, surface-based investigations is sufficient for the assessment of long-term safety. The site has favourable conditions for safety and no site-related issues requiring resolution in order to demonstrate safety have been identified. Confidence in the site-descriptive model and in the understanding of the site is obtained by a systematic and quality assured programme for site investigations and site modelling. The confidence in the site model is assessed in detail and documented in the site description Forsmark report (SKB, 2008).
- The reference design is well-established, with specified and achievable production and control procedures yielding an initial state of the repository system with properties favourable for long-term safety at the Forsmark site. The engineered parts of the repository system are based on demonstrated technology and established quality assurance procedures to achieve the initial state of the system. There is potential for additional optimisation when this reference design is developed and implemented.
- The scientific understanding of issues relevant for long-term safety is mature as a result of decades of research both within Swedish and other national programmes and in international collaboration projects. The R&D efforts to understand repository evolution and safety have led to the understanding of key processes like copper corrosion, shearing of canisters and other potential canister failure causes, and of key phenomena controlling retardation. This knowledge is, in SR-Site, systematically documented in several reports in a format suitable for use in the safety assessment (SKB, 2010b, 2010c, 2010k, 2010l).
- The SR-Site main report and its supporting documents have been comprehensively peer reviewed. In particular the scientific basis of the safety assessment has been reviewed by recognised experts in the relevant scientific fields.
- A complete analysis of issues identified as relevant to long-term safety (SKB, 2010j) was carried out for the SR-Site according to an established assessment methodology (SKB, 2011), comprising e.g. cautious approaches when addressing uncertainties:
  - The understanding of safety is built on a systematic identification of safety functions and criteria for the safety functions.
  - Repository evolution is analysed with a structured approach in several time frames, addressing in each of these the processes that have been identified as relevant and with the safety of the system, as expressed by the safety functions, as a focus. Data uncertainties and data quality are assessed and documented according to a pre-established template (SKB, 2010d). Model and modelling QA is achieved by following procedures documented in a model summary report (SKB, 2010n). The assessment is then broken down into a set of scenarios to exhaustively scrutinise all possible ways in which the identified safety functions could be impaired and the potential consequences of such situations.

  - Confidence in the key results of radionuclide transport and risk calculations is enhanced by the fact that they can often be closely reproduced with simple, analytical models, using the same input data as the fully qualified numerical models.
  - The key results of radionuclide transport and risk calculations are overestimates, since a number of pessimistic assumptions were made in the analyses, both regarding the extent of canister failures and regarding their consequences.
- Documented quality assurance routines have been applied in the assessment of the initial state, in the development of the site description and in the analysis of long-term safety. A QA plan, encompassing most of the routines followed in undertaking the steps described in the above points, has been established and implemented at the SR-Site. This is part of the overall methodology followed in the assessment.

Freedom for later decisions

In addition to the scientific input regarding the future evolution of the repository barriers, the safety case rests on the design of the repository barriers and the layout adaptation to the bedrock conditions. In developing the design and layout of the repository, designers need to combine several requirements and preferences with respect to capacity and functionality, long-term safety and the environment. For these reasons, and in accordance with regulations, SKB has developed design premises in terms of requirements and other conditions and presented these to the designer (SKB, 2009). The design premises typically concern specification on what mechanical loads the barriers must withstand, restrictions on the composition of barrier materials or acceptance criteria for the various underground excavations. A reference design has been developed and reported in a series of production reports (SKB, 2010e, 2010f, 2010g, 2010h, 2010i, 2010p). The production reports state the design premises to be fulfilled, present the selected reference design, present verifying analyses that the reference design conforms to the design premises and outlines production and control procedures selected to achieve the reference design with verifying analyses that these procedures do achieve the reference design together with an account of the achieved initial state. The design, the design premises and the scientific basis for the safety case may need to change in later stages of the repository detailed design work, construction and operation. As a potential license obtained based on the currently submitted license application will be based on statements and facts of the currently submitted safety case, such changes can only be made in an orderly fashion.

According to Swedish regulations SSM FS 2008:1 (Chapter 4, Section 1):

A preliminary safety analysis report shall be drawn up before a facility may be constructed and, for an existing facility, before major refurbishing or rebuilding work or major modifications are carried out. The safety analysis report shall be updated before trial operation of the facility may commence so that the report reflects the construction of the facility.

Furthermore, the regulation SSM FS 2008:1 (Chapter 4, Section 5) states:

Technical and organisational modifications to a facility, which can affect the conditions specified in the safety analysis report, as well as principal modifications in the safety analysis report, shall be subject to a safety review in accordance with Section 3. Before modifications in accordance with the first paragraph may be implemented, the Swedish Radiation Safety Authority shall be notified of the modifications.

According to SSM FS 2008:1 (Chapter 4, Section 3):

A safety review in accordance with the provisions of these regulations shall be performed in order to verify that applicable safety aspects have been taken into account and that applicable safety requirements with respect to the design, performance, organisation and activities of the facility are met. The review shall be performed in a comprehensive and systematic manner and shall be documented. The safety review shall be performed in two stages. The first stage, the primary review, shall be performed within the parts of the facility's organisation that are responsible for the specific issue. The second stage, the independent safety review, shall be performed within a safety review function appointed for this purpose, which shall have an independent position relative to the parts of the organisation responsible for the specific issue.

The following is noted:

- SKB retains the freedom to explore and develop design changes or other changes to the safety case.
- All potential changes need to be assessed with regard to their impact on safety and whether the changes lie within the scope of the safety report (e.g. the safety case).
- The assessment needs to be reviewed in accordance with the rules set out in SSM FS 2008:1.
- Before changes are actually implemented, SSM shall be notified of the modifications and the assessment of its implication.

Before receiving on operational license, SKB would not formally need an approval by SSM, but such changes would be made at SKB's own risk, since SSM might later reject the implementation of the change or ask for more information, including revision of the safety report, before accepting implementation of the change. Changes that could imply a significant impact on the safety case would need relatively more preparation and justification, compared to changes relating the details within the framework of the design and information presented in the license application.

Planning for construction, operation and closure

While SKB has established a technically feasible reference design and layout of the KBS-3 repository and showed that this conforms to stated design premises, technical development will continue. Detailed designs adapted to an industrialised process fulfilling specific requirements on quality, cost and efficiency need to be developed. The layout also needs to be adapted to the local conditions found when constructing the repository at depth. In accordance with the scope of the safety case presented in the license application, these potentially more optimal solutions should result in at least the same level of safety as the reference design assessed in the SR-Site. A strategic technology development plan has been established, covering the remaining development work from now until the repository is constructed, all technical systems are implemented and operation can start. Major milestones in the KBS-3 system development and what needs to be accomplished at these milestones have been identified (SKB, 2013). Some important examples are given below.

Before start of construction of an encapsulation plant and the final repository the canister detailed design needs to be completed, verified against the design premises and it should be shown that selected means of canister production and quality control will allow assessment of conformity with the design premises. Buffer and backfill system design should be completed and include a preliminary quality control plan for the production of the bentonite blocks and the installation in the deposition tunnels. A detailed design of the repository underground access with descriptions of the requirements, methodology, execution and results verification (authentication) for all underground construction activities in the repository is needed. For the deposition areas, a general plan is needed showing how the underground construction work and the detailed investigations will lead to selection of deposition tunnels and holes in conformity with the set design premises. These descriptions need to give a sufficiently clear picture of the work and the end result, but the development of detail can continue until the detailed design of the deposition area starts.

As a basis for the detailed design of the production facility for bentonite blocks the detailed design of the buffer and the backfill needs to be completed. This includes setting requirement specifications for the bentonite materials, decisions on press technology for buffer blocks and establishing manufacturing and quality control methods that work on an industrial scale for the production.

Deposition tunnels will only be excavated when needed for deposition, and their excavation is generally seen as part of repository operation. A final decision on location of deposition tunnels and deposition holes will also be decided at this point, based on the outcome of detailed investigations and pre-set plans for acceptance or rejection of tunnels and holes, using the basic approach of the “observational method”. However, the basic procedure for this operation, as well as the detailed design of the first repository tunnels, needs to be established earlier. Installation methods and methods for testing and control of the buffer and backfill must be designed in detail and verified, requiring full-scale underground testing. The overall site understanding should be updated with the investigation data obtained from underground, the application of criteria for selection of deposition holes tested and the suggested technologies for underground construction of deposition tunnels and drilling deposition holes should be shown to work under the specific underground conditions at Forsmark.

Before the safety report can be updated and the operational license application submitted, commissioning tests of the entire KBS-3 system, ranging from canister manufacturing and covering all steps needed until a deposition tunnel is backfilled and plugged, are required. Before such commissioning tests can be undertaken different types of integration tests to ensure that the equipment and technological systems work together as intended would be needed, followed by modifications if necessary.

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SSM's licensing review of a spent nuclear fuel repository in Sweden

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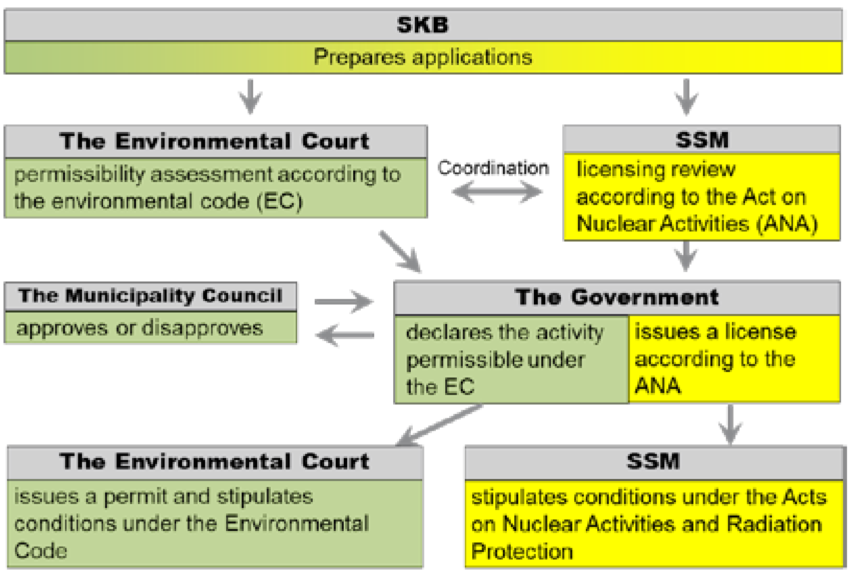
Introduction

On 16 March 2011 the Swedish Nuclear Fuel and Waste Management Co. (SKB) submitted license applications for a general license to construct, possess and operate a KBS-3 type spent nuclear fuel repository at the Forsmark site, in Östhammar municipality, and an encapsulation plant in Oskarshamn municipality. The KBS-3 method, which has been developed by SKB over a period of more than 30 years, entails disposing of the spent fuel in copper canisters, surrounded by a swelling bentonite clay, at about 500 m depth in crystalline basement rock.

SKB's applications are being evaluated in parallel by the Swedish Radiation Safety Authority (SSM) according to the Act on Nuclear Activities and by the Land and Environmental Court according to the Environmental Code (see Figure 1). During the review SSM will act as an expert review body to the Land and Environmental Court in the areas of radiation protection, safety and security/non-proliferation. Both SSM and the court will produce a statement with a recommendation regarding a licensing decision and licensing conditions to the government. The government will make the final decision after consulting the municipalities concerned by SKB's facilities (municipal veto applies).

Figure 1: The licensing review procedure



The current licensing decision is just one of several licensing decisions that will be required for the repository. However it is arguably the most important one, because it is the last licensing stage with a broad societal involvement including an environmental impact assessment (EIA) process, national consultations and municipal veto for the concerned municipalities. The licensing steps to follow, should SKB be granted a license by the government, only require approval by SSM. These steps include application for start of actual construction work, test operation and routine operation.

Evolution of review approach and staff training

SSM and its predecessors [the Swedish Nuclear Power Inspectorate (SKI) and the Swedish Radiation Protection Institute (SSI)] have for almost 30 years acquired review experience and knowledge about the KBS-3 disposal method through repeated reviews of SKB's preliminary safety assessments for the KBS-3 disposal method (SKI, 1984, 1992; SKI/SSI, 2001, 2008). In parallel, Swedish regulators have also carried out an ambitious independent research programme aiming at the development of an independent safety assessment capability, including two safety assessments of the KBS-3 disposal method (SKI, 1991, 1996).

The main problem with staff training is the very long time scales of the Swedish repository project and the fact that key milestones often have been put forward in time. Only a few of the staff members who were involved in the training activities are still with SSM; in fact, licensing review is carried out mainly by newly employed staff. In retrospect it can be concluded that more thought should have been given to the development of a strategy for maintaining review competence and knowledge management. There are also important differences between reviewing a preliminary safety assessment and a licensing review. The early reviews focused on identification of weaknesses and providing general recommendations on improvements. While the early phase of the current licensing review has indeed been quite similar, SSM now also has to complete a comprehensive issues resolution phase with detailed consideration of applicable regulations and guidelines. In the past, no such definitive compliance judgments have been made for the KBS-3 disposal method. Finally, it should be recognised that SKB's safety case for the KBS-3 method has evolved considerably over the last couple of decades. Where the quantitative part of the analysis previously focused heavily on dose and risk associated with a failed canister, there is now more focus on complex quantitative analyses of the containment safety function.

Pre-licensing interactions

As early as the 1980s, Swedish regulators as well as the Swedish government have had opportunities to comment on and influence SKB's programme through their research, demonstration and development (RD&D) programmes, which are published every third year. The fact that SSM and its predecessors have had a formal role to play even before licensing, has contributed to SSM's current knowledge about SKB's programme. Not entirely unproblematic is the biased knowledge created by this process with extensive information on the KBS-3 disposal method, but only limited knowledge about alternative concepts. As explained later in this paper, this is an issue because method selection is within the scope of the ongoing licensing review.

Between 2001 and 2008 SKB carried out site investigations at the Forsmark and Laxemar sites. During this period SKB had formal consultation meetings with SKI and SSI targeting both SKB's site investigations and safety assessment work (SKB, 1999, 2006) in which site investigation results were utilised. The strong commitment to these meetings from both sides is partially explained by the fact that they were required by a government decision. During this period SKI and SSI also carried out detailed technical and scientific reviews of site measurements and modelling work with the help of external experts. The outcome of these reviews was regularly communicated with SKB to make sure that they had a chance to provide the information expected by SSM at the time of licensing. This work has considerably facilitated the ongoing licensing review, since SSM would otherwise have had to devote much more attention to site-specific measurements and modelling work in support of the current safety assessment. However, it is important to note that SSM has not formally approved SKB's site investigations and can thus question any aspect of the site investigations if new circumstances or review contexts were to emerge in the ongoing review of the SR-Site safety assessment (SKB, 2011).

Review objectives and goals for the review process

The overall objective of SSM's licensing review is to develop a scientifically-based review statement and a recommendation for a government decision on SKB's license applications. Given the current stage of the licensing process the following review criteria need to be addressed:

- Has SKB made a convincing argument for compliance with SSM's regulatory requirements on operational and post-closure safety? The final compliance judgement on post-closure safety cannot be made until SKB has been given permission to carry out underground detailed site investigations and confirm the conditions at depth. At the current stage SKB should be able to demonstrate that there are no unresolved safety critical technical or scientific issues.
- Has SKB demonstrated technical feasibility of the KBS-3 method? Although some technical development remains for the implementation of repository construction and operation, there should not be any remaining uncertainty as to whether the repository system can be constructed and operated so as to achieve the initial state assumed in the safety analysis.
- Closely related to the above question is whether SKB has presented credible plans for the handling of the remaining technical and scientific issues. This includes plans for verification of methods for full-scale manufacturing and testing of engineered barrier components, demonstration and verification of the full sequence of canister emplacement and backfilling of tunnels and any plans for long-term experiments/ monitoring aiming at confirming safety functions.
- Has SKB justified the selection of disposal method and site in relation to alternatives? This is an important criterion at this stage, because it is now that the selection of both the disposal method and the site is made.

SSM developed the review plan with the view of honouring its three value words: trustworthiness, integrity and openness. Trustworthiness is something that has to be earned by the observer. However, as mentioned above, SSM (and its predecessors) have prepared for this licensing review by reviewing SKB's development work and by developing an independent competence in safety assessment for more than 25 years. Integrity involves maintaining SSM's independence of SKB and the nuclear industry but also avoiding being unduly influenced by any external stakeholder. To achieve this SSM has defined strict criteria for the selection of external experts, both competence criteria and criteria concerning conflicts of interest. SSM has also defined rules for all types of interaction with SKB during the review. Several measures were taken to achieve openness and transparency of SSM's review work as described in the section on stakeholder interaction below.

SSM's review organisation

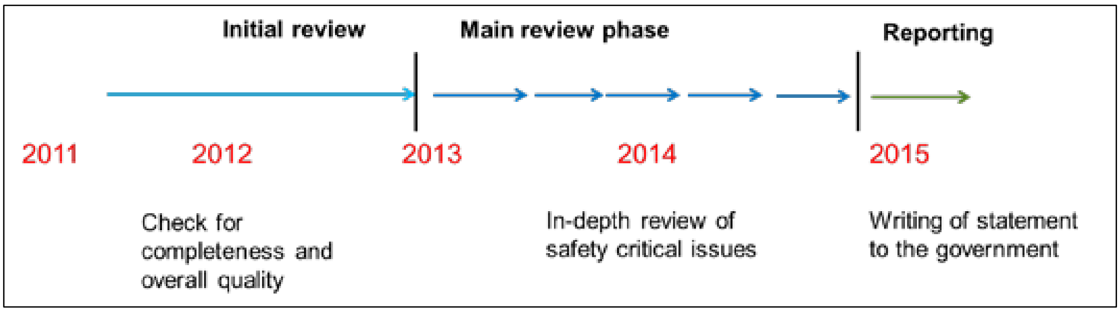
SSM's licensing review is organised as an authority project, owned by SSM's director general. The operational review work is carried out in four sub-projects: *i)* operational safety of the repository; *ii)* post-closure safety of the repository; *iii)* the encapsulation plant; *iv)* repository system issues. Almost 40 SSM staff are involved to some extent in the review; the total amount of staff resources spent on the project is about 12 person years per year. SSM's licensing review is fully funded by the Swedish Nuclear Waste Fund.

Neither SSM nor its predecessors have had access to a dedicated technical support organisation. External expert support has instead been provided through the development of long-term working relationships with external experts funded through research programmes available to the Swedish regulators. However, as the Swedish programme has now moved into a licensing phase external expert support can no longer be regarded as research, which implies that the Swedish Public Procurement Act becomes applicable. Therefore SSM had to organise an extensive public procurement of external expert support. Strict impartiality requirements were defined to ensure independence from SKB and to maintain the integrity of SSM's licensing review. All in all almost 40 international experts received three-year framework agreements in one or more of 17 scientific and technical review areas. The whole process of public procurement took more than a year and was quite demanding in terms of resources. However, it is believed that the use of an unbiased and open procurement procedure for the procurement of SSM's external experts' support is essential for the trustworthiness of the review process.

Review method and scope of SSM's licensing review

SSM's licensing review is divided into three phases: the initial review phase (IRP), the main review phase and the associated reporting phase (see Figure 2). The IRP started with SKB's submission of their license applications in March 2011 and was completed at the end of 2012.

Figure 2: Schedule for SSM's licensing review



The objectives of the initial review phase were to make a broad review of all primary licensing documents of SKB's safety case in order to make a first assessment of the quality and completeness of SKB's application, to identify scientific and technical areas for an in-depth review in the coming main review phase, and to develop requests for complementary information (RCIs) from SKB. The initial review phase comprised the following activities (some of which are elaborated below):

- public procurement of external experts;
- document review by SSM staff and external experts covering SR-Site and the primary supporting references;
- first round of SSM's independent modelling;

- first round of a national consultation;
- international peer review of SKB's license application;
- development of requests for complementary information (RCI);
- development of a statement on completeness of SKB's applications to the Land and Environmental Court.

SSM's overall conclusion from the initial review phase was that SKB's reporting is sufficiently comprehensive and of sufficient quality to justify a continuation of SSM's review. However, as described in more detail below, SSM has identified a number of both technical and scientific issues requiring complementary information from SKB, for the in-depth review in the main review phase. There is now a range of review results that can be accessed through SSM's website (www.ssm.se) including: 35 technical notes that document the review findings of SSM's external experts, a large number of requests for complementary information and answers from SKB, review comments from the national consultation, the final report of the international peer review organised by OECD/NEA (2012) and the statement to the Land and Environmental Court on completeness of SKB's applications (only available in Swedish).

The main review phase that started in January 2013 involves an in-depth review and resolution of the safety critical review issues identified in the initial review phase. It will continue until all review issues of importance for the compliance evaluation have been resolved and will be completed with a preliminary compliance statement to the Land and Environmental Court in early 2015 for their main hearing. The final review phase, currently scheduled for the first half of 2015, will be devoted to completing all review documentation and finalising the review statement to the government. The main review and reporting phases comprise the following activities:

- structured resolution of remaining review issues based on complementary information from SKB and support from SSM's external experts;
- second round of SSM's independent modelling;
- second round of the national consultation;
- development of a second review statement to the Land and Environmental Court;
- development of a review statement with a recommendation for a licensing decision to the Swedish government.

Independent modelling

SSM and its predecessors have a long tradition of using in-house independent modelling as a review tool. The experiences from previous reviews of SKB's preliminary safety assessments show that reproducing part of SKB's calculations improves the understanding of SKB's analysis and provides insight into quality assurance problems that are hard to identify only from document reviews. The first round of independent modelling that was carried out during the initial review phase aimed at reproducing SKB's consequence calculations (radionuclide release, transport, dose and risk calculations). Through this modelling SSM has identified a number of review issues requiring either complementary information from SKB or further in-depth review, e.g. inconsistency between documentation and actual modelling performed in SKB's dose assessment, QA problems and insufficient justification of data and assumptions. During the main review phase a second round of independent modelling will be carried out with the aim of addressing key conceptual model uncertainty using alternative conceptual models. This part will include both in-house models for the consequence analysis and selected process modelling using models through SSM's external experts.

National consultation of SKB's license application

In order to promote openness and broad public participation in the licensing review, SSM has organised a national consultation of SKB's license applications in two rounds. In the first round, carried out during the initial review phase, SSM requested views primarily on overall quality and completeness of SKB's applications. The applications were sent to a total of 67 organisations including the involved municipalities and county boards; environmental organisations and other non-governmental organisations; universities and other authorities.

SSM received a broad range of review comments covering most aspects of SKB's safety case, including safety analysis methods. Important areas of concern include the completeness of SKB's environmental impact statement and SKB's justification of the selected repository site and the KBS-3 disposal concept. A common view was that the very deep borehole disposal method should be more thoroughly investigated. SSM also received rather detailed technical review comments concerning the long-term integrity of the copper canister and in particular copper corrosion mechanisms. The review comments have been compiled and taken into account by SSM in developing its statement on completeness and overall quality of SKB's applications after the initial review phase. A second round of national consultation will be carried out during the main review phase with a focus on safety critical issues in SKB's application.

International peer review

Another external input to SSM's review is the international peer review of SKB's post-closure safety report, SR-Site (SKB, 2011), which was organised by the OECD/NEA upon request from the Swedish government. The remit of the peer review was to provide a statement, from an international perspective, on the sufficiency and credibility of SKB's post-closure radiological safety case for the licensing decision at hand. The peer review was carried out according to the established format for sponsored NEA peer reviews (OECD/NEA, 2005a, 2005b) with written questions to SKB and a hearing of SKB halfway through the peer review. The final review report (OECD/NEA, 2012) was presented to SSM and other stakeholders in June 2012.

The overall conclusion of the peer review is that SKB's post-closure safety report is sufficient and credible for the decision at hand and that SKB generally gives a convincing illustration and technical basis both for the feasibility of the future repository, according to the KBS-3 design, and for its radiological long-term safety. However, the expert team also gave a number of recommendations for additional research and improvements that are needed for the safety cases supporting the next licensing steps. The team also underscored that the progression from the conceptual phase of SKB's repository project to an implementation phase means that the industrial feasibility of the barriers and of the repository, including assurance of their quality, will become increasingly important.

The Swedish regulatory authorities have a long tradition of using the NEA services for organising international peer reviews in connection with reviews of SKB's safety assessments, but also to get a quality assessment of the authorities' independent safety assessments (OECD/NEA, 2000; SKI, 1997). In SSM's view, the international peer review of SKB's post-closure safety report has an important role in adding to the trustworthiness of the review process. NEA's long experience of developing and reviewing safety cases warrants a qualified opinion on SKB's application *vis-á-vis* the international state of the art. The broad representation of international experts in the international review team also made it possible for the team to identify a number technical review issues in support of SSM's and other stakeholders' review. Finally, the fact that the international peer review was carried out independently in relation to SSM's review also means that it provides a second opinion for the municipal (the veto decision) and governmental decision makers.

Stakeholder interaction

Sweden has a long tradition of openness and stakeholder interaction in the area of nuclear waste management. The need for a an active stakeholder interaction became apparent in the early 1990s when SKB started their national siting process for a spent nuclear fuel repository and the municipalities involved started to develop their own competence concerning these issues. Since then the Swedish regulatory authorities have developed methods for stakeholder dialogue with the municipalities involved and non-governmental organisations (NGO). As a result, the authorities' communication strategy gradually changed from one-way information to an active dialogue and support of the municipalities and NGO. Today the municipalities and some of the NGO involved in the licensing review have developed an independent expertise and make their own assessment of SKB's applications.

Several measures have been implemented to stimulate and facilitate broad societal involvement in the ongoing licensing review, including the national consultations of SKB's applications mentioned above. SSM also strives to be as open as possible in its review and in making interim results available to the public. SSM's website has been developed to provide background information on the licensing review as well as updated information on the progress of SSM's review. Information meetings are organised when review milestones have been achieved, for example after the completion of the international peer review and in connection with the national consultations.

Preliminary review findings and requests for additional information

Although it would be difficult for SSM to go public with preliminary judgments of SKB's application during the review, SSM can inform about what it thinks are important review issues and upon what topics it has requested complementary information from SKB. The following categories give a broad picture of the type of information that SSM has requested thus far during the initial review phase:

- selection of site and disposal method;
- long-term canister integrity;
- site-specific conditions at Forsmark;
- plans for technical implementation and resolution of remaining issues;

These areas of request for complementary information fairly well match the areas for in-depth review in the ongoing main review phase.

Selection of site and disposal method

Despite SKB's comprehensive and long running research, development and demonstration programmes for the KBS-3 concept, no formal decision has been made in Sweden regarding either the acceptability of this method or the selection of the Forsmark site. Consequently, SSM's licensing review has to address SKB's justification of the KBS-3 disposal method and the Forsmark site. This involves comparison with entirely different disposal methods and waste management options as well as alternative sites. Given the fairly advanced stage of SKB's KBS-3 programme, it is quite difficult to set and justify boundaries on how far reaching such considerations would have to be. So far, SSM has requested a more comprehensive comparison of the mined KBS-3 disposal method with the fundamentally different disposal concept of very deep (3-5 km) bore holes, taking into account recent technical developments in the area of deep drilling in basement rocks. SSM has also requested SKB to elaborate on the possibility of using the spent nuclear fuel as a resource in the future. Such reasoning is needed in order to justify direct disposal in relation to the requirements in the Environmental Code on recycling and a cautious use of energy resources. Regarding the justification of site selection, SSM needs to consider requirements both in the environmental legislation and in SSM regulations. One specific issue where SSM has requested complementary information is the justification of a coastal site like Forsmark in relation to an inland recharge area site with potentially longer transport paths to the biosphere and a different future evolution of groundwater salinity.

Long-term canister integrity

SKB's post-closure safety case relies predominantly on the containment safety function. First, SKB assumes that out of the 6 000 canisters to be manufactured and sealed, none will have an initial defect that will significantly affect their long-term performance. Second, SKB's modelling work in SR-Site suggests that canister failure will only occur to a very limited extent and mostly far into the future. SKB has identified two viable canister failure modes: *i)* copper sulphide corrosion subsequent to buffer erosion due to glacial meltwater intrusion into deposition holes; *ii)* shear-failure of a canister caused by a fracture shearing exceeding 5 cm across a deposition hole originating from a large earthquake in the vicinity of the repository. The results of the analyses suggest that for both failure modes the probability of failure would be less than that corresponding to a single canister failure over the entire assessment period of 1 million years. It should be noted that SKB's analysis also contains calculations covering cases with much more extensive degradation of engineered barriers, but overall SKB emphasises the containment safety function more than the retardation safety function.

SSM has devoted considerable review resources to scrutinise SKB's analysis of the containment safety function and has requested complementary information related to the two failure modes described above. For the purpose of analysing canister integrity SSM has also focused on other conceivable reasons why canisters may fail and in particular may fail at early time frames in the safety assessment. Two safety critical failure modes, for which SKB does not expect any canister failures, are brittle creep deformation associated with the closing of the initial gap between canister shell and the canister insert, and the influences of high isostatic external pressure on the canister due to the hydrostatic water pressure and swelling pressure from the buffer. These two conditions are applicable to all canisters in the repository as opposed to only the most exposed canisters in SKB's two anticipated failure modes. According to SSM's judgment uncertainty related to potential common-mode failure mechanisms has to be small since they are associated with potentially considerable consequences. SSM's requests for complementary information related to canister corrosion include:

- justification for eliminating localised corrosion processes, such as pitting and stress corrosion cracking (early canister corrosion failure would require a localised corrosion process);
- quantification of general corrosion, e.g. assumptions regarding microbial sulphate reduction and the role of anoxic copper corrosion;
- quantitative description of buffer erosion including canister area exposed to groundwater flow;
- long-term evolution of groundwater salinity and number of deposition holes affected by buffer erosion processes.

SSM's requests for complementary information related to canister mechanical properties include:

- the basis for ruling out creep brittle failure and the role of phosphorus for the creep ductility of copper;
- understanding of mechanisms involved in copper embrittlement.

Some of SSM's information requests have required SKB to conduct completely new research work, while other requests require clarifications and complementary calculations.

Site specific conditions at Forsmark

The Forsmark candidate site is a tectonic lens surrounded by major deformation zones with the main volume at repository depth comprising of rock with unusually few fractures and water conducting features. Since such tight rock conditions have not been fully addressed in the past, the nature of performance issues related to both engineering feasibility and long-term safety must be scrutinised. SSM has for instance requested information related to the initial and long-term evolution of rock stresses which appear to be higher than for normal Swedish bedrock. Furthermore, tight rock volumes surrounding deposition tunnels and deposition holes imply that the time period of unsaturated conditions in the clay material and bedrock may be more extended than previously envisaged, according to SKB several thousands of years. SSM has requested that SKB evaluate associated uncertainties. Preservation of unsaturated conditions in the clay material for a long time on the one hand reflects favourable slow transport conditions in the bedrock, but on the other hand such conditions are not fully analysed and are not fully compatible with the ideal state of the engineered barriers.

In the SR-Site safety assessment, SKB emphasises geosphere performance of the Forsmark rock at repository depth in the context of ensuring canister integrity. The sealing ability of the buffer is on the contrary downplayed by accounting for buffer erosion in a number of deposition holes and by including a pessimistic account of buffer erosion which has taken place in all deposition holes. SKB's safety case is therefore considerably dependent on detailed site-specific conditions. Realisation of the expected good geosphere performance assumes that SKB has the ability to detect and avoid positions in the bedrock susceptible to large mass transfer and extensive shear movement. SSM consequently focuses on scrutinising SKB's plans to confirm geosphere performance and further evaluating phenomena that can disrupt tight rock volumes.

Plans for technical implementation and resolution of remaining issues

The current licensing step involves a general license for construction, possession and operation of a spent nuclear fuel repository at the Forsmark site. However, if SKB were to be granted a government license they would need to submit a preliminary safety analysis report and receive a permit from SSM before they could go underground with tunnels and shafts and start the actual construction work at the site. Additional permits will be needed prior to both test operation and routine operation. Because all aspects of the disposal method cannot be resolved in the current licensing stage SSM needs to determine what is necessary to know now and what can wait until later. SSM will need at least preliminary plans for those aspects that will be addressed in future steps of the programme. SSM has thus far identified a need for complementary information regarding plans related to the following aspects of SKB's programme: *i)* plans for demonstration and verification of canister and engineered barrier emplacement to be conducted prior to repository operation; *ii)* plans for optimisation of repository depth; *iii)* plans for performance confirmation involving measurements and long-term experiments related to engineered barrier performance and site conditions; *iv)* plans for development and refinement of the design basis cases and design premises.

Discussion

The completion of the initial review phase marks a first milestone in SSM's licensing review, which has given SSM a good picture of the overall quality of SKB's application and the need for complementary information. SSM has now moved into the main review phase with an in-depth review of safety critical issues. Because this licensing review is a first of its kind, SSM is continuously learning and updating its review plans. Some observations that can be made so far are:

- Although there is a broad international consensus on the merits of a stepwise process for a repository programme, it leads to some questions regarding what information is sufficient for different licensing steps. In spite of the anticipated arrival of important complementary information from SKB, there will be remaining uncertainty concerning both repository implementation and long-term safety. SSM will need to decide whether such uncertainty is of a character that can jeopardise the overall safety objectives for a KBS-3 repository at the Forsmark site, or if it can be handled at later stages of the SKB programme through license conditions or future RD&D programmes.
- The very long time scales associated with a repository programme require a good strategy for maintaining review competence. Although SSM has prepared for this licensing review for many years, the benefits from these preparations were not as substantial as expected, mainly due to the lack of a systematic knowledge management.
- The fact that Sweden has not made a decision-in-principle concerning a waste management option and site selection leads to a very broad review context and it has proved to be a challenge to handle the wide array of review issues ranging from scientifically detailed ones associated with the proposed KBS-3 method, to general issues like the justification of the selection of site and disposal method.
- A licensing review involves new types of compliance judgments, which are difficult to prepare for beforehand; a systematic approach for decision making and issue resolution is needed.
- SKB's post-closure safety case underpinning the license application has evolved from earlier preliminary safety assessments and now contains more detailed analyses of the containment safety function. This change has forced SSM to adjust its review focus accordingly.
- The broad societal involvement in the licensing review puts high demands on SSM's interaction with stakeholders. SSM has strived for adopting the review process according to its value words: trustworthiness, integrity and openness.

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Session 3.6

Safety Case for License Application for a Final Repository: The Finnish Case

TURVA-2012 safety case for licensing a spent fuel repository at Olkiluoto, Finland

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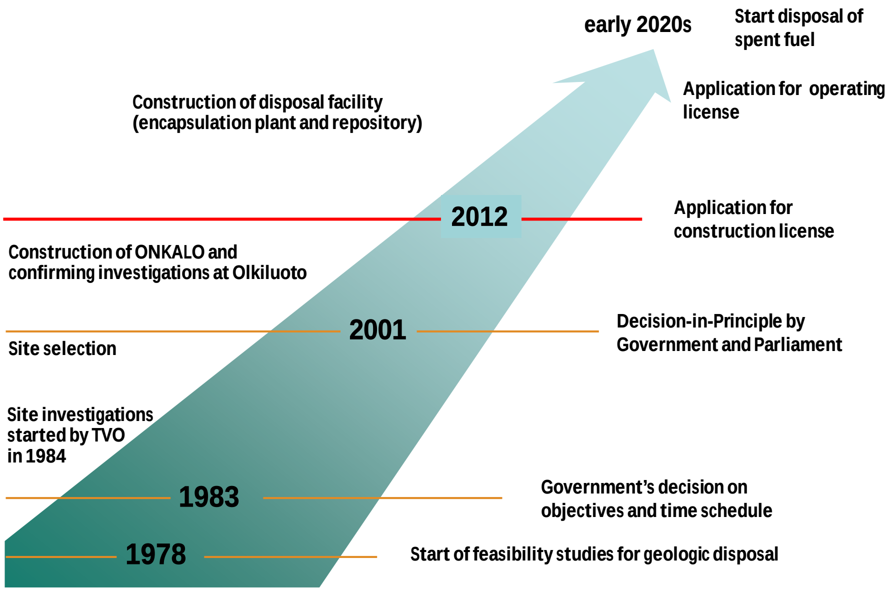
Programme status

In 2001, the Finnish Parliament endorsed a decision-in-principle (DiP) whereby the spent nuclear fuel produced by the operating nuclear reactors at Olkiluoto and Loviisa will be disposed of in a geological repository at Olkiluoto, on the south-western coast of Finland (Figure 1). Subsequently, additional DiPs were issued allowing the extension of the repository to accommodate spent nuclear fuel from additional reactors that are under construction or in planning at Olkiluoto, which means a total of 9 000 tU of spent nuclear fuel to be disposed of. In accordance with the decision of the Ministry of Trade and Industry (KTM) in 2003, Posiva submitted an application for a license to construct a disposal facility at Olkiluoto in 2012, consisting of an encapsulation facility and an underground deep geological repository. The application included a Preliminary Safety Analysis Report (PSAR) and a long-term safety case, TURVA-2012. Assuming a positive outcome of the current licensing review, the next step would be the Final Safety Analysis Report (FSAR) in support of an operational licence application around 2020. A general time line for Posiva's programme is presented in Figure 2.

Figure 1: Olkiluoto Island is situated on the coast of the Baltic Sea in south-western Finland

Figure 2: Time line for spent fuel disposal of the Loviisa and Olkiluoto reactors; the target is to start disposing of spent nuclear fuel in the early 2020s



The disposal method is based on the same KBS-3 concept that the Swedish SKB has used as basis for their license application in 2010. Accordingly, the spent nuclear fuel will be encapsulated in water- and gas-tight copper canisters equipped with a load-bearing insert and emplaced in a deep geological repository constructed in the bedrock. The canisters will be surrounded by a swelling clay buffer material that isolates them from the bedrock. The deposition tunnels and the central tunnels and the other underground openings will be backfilled with materials of low permeability. The repository will be at a depth of about 400-450 m below ground. The primary role of the bedrock is to provide sufficiently stable conditions for the engineered barrier system and to make inadvertent human intrusion unlikely. In case of EBS failure, the bedrock shall also retain and retard the possible spreading of radionuclides from the repository.

Site

Olkiluoto Island has been investigated for siting of nuclear waste repositories for over 25 years; first for siting of a low- and intermediate-level waste repository, later for a deep geological repository for spent nuclear fuel. By 2012 a total of 57 deep boreholes had been core-drilled on the site. Since 2004 the Olkiluoto bedrock has also been subject to underground investigations from the ONKALO underground rock characterisation facility.

Underground characterisation before the application of the construction license is required by Finnish regulations to confirm the results of the surface investigations carried out for the basis of the decision-in-principle. Indeed, the underground investigations have indicated that the structural description of the Olkiluoto bedrock based on the surface investigations was largely correct and the main structural features were found where they were expected. In general, there were hardly any surprises in the geological picture as unveiled. However, some of the bedrock characteristics could only be confirmed with the ONKALO studies. One example is the rock mechanical description, which turned out to be a major challenge before going underground. The picture of fracturing also benefitted considerably from the ONKALO investigations: while the larger hydraulic features were quite well predicted from the surface, the more detailed nature of fracturing could only be determined through underground investigations. For instance, the number of "large" single fractures turned out to be significantly larger than what was predicted on the basis of surface studies. Still another feature that emerged from the underground investigations was the matrix geochemistry. The results of the matrix water investigations are still partly open to discussion.

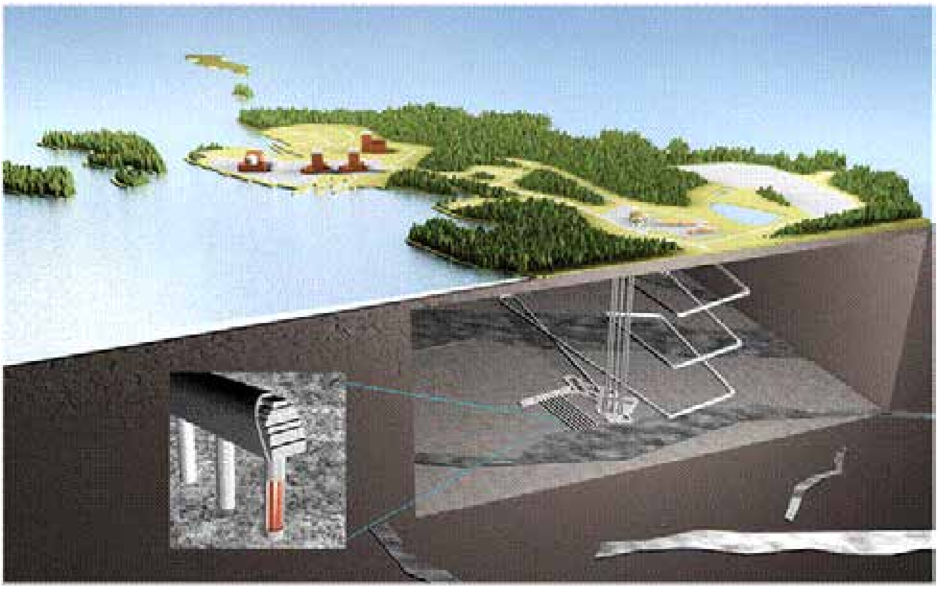
For the future understanding of the evolution of the Olkiluoto bedrock conditions the key features, events and processes taken into account in the assessments of long-term performance and safety include:

- the presence of deformation and fractured zones, displaying mixed geotechnical properties and in some cases increased hydraulic activity;
- higher rock stress at depth, which may cause disturbance to the rock, making underground openings less stable;
- the temperature and thermal conductivity of rock and residual heat output of the spent nuclear fuel;
- the high salinity of groundwater at depth, which may affect the performance of the engineered barriers;
- continuing post-glacial crustal uplift and, in the longer term, climatic cooling and glaciation, leading to changes in rock stress and potential changes in groundwater flow and composition, e.g. influx of dilute glacial melt waters into the host rock.

Design approach

Posiva has developed a detailed design for the Olkiluoto repository (Figure 3) through a formal requirements management system (VAHA). This provides a rigorous, traceable method of translating safety principles and the safety concept to a set of safety functions, performance requirements (performance targets, target properties), design requirements and design specifications for the various barriers, i.e. a specification for the implementation of the disposal concept at the Olkiluoto site.

Figure 3: Schematic illustration of the underground repository features and of the (vertical) KBS-3 design (insert)



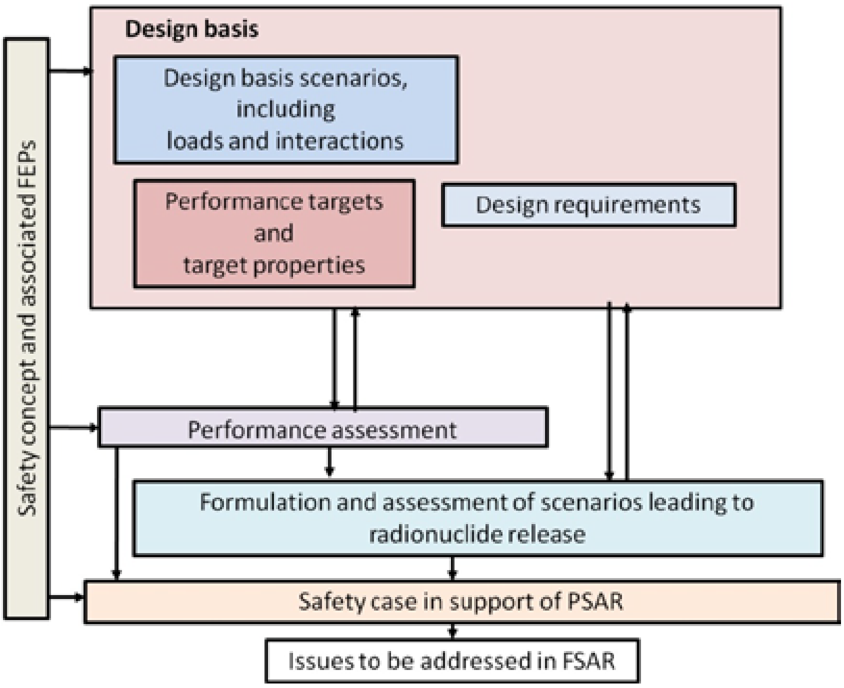
Posiva's safety concept is based on two basic principles for safe spent nuclear fuel disposal – isolation and containment – provided by a system of mutually complementary natural and engineered barriers and their associated safety functions. In this way, even if the performance of one of the barriers is degraded in some way, other barriers and safety functions will still ensure that isolation and containment of the spent nuclear fuel are maintained. The system of complementary barriers and safety functions also provides robustness with respect to external events and processes, including geological and climatic changes. The conditions needed for the barriers to fulfil their respective safety functions are expressed in terms of performance targets for the engineered barriers and target properties for the host rock.

The performance targets and target properties were then transformed to design requirements for engineered barriers and the definition of a rock suitability classification (RSC) system, by which the local suitability of the rock for deposition tunnels and holes can be assessed.

Figure 4 describes the relationships between the development of the design basis and the formulation of the safety case: the design basis is developed iteratively on the basis of performance assessment and the formulation and assessment of radionuclide releases. At each stage, the available scientific understanding, including the results from earlier assessments, is used to define the current design basis; it may, however, need to be modified as a response to the outcome of the performance and safety assessment.

Figure 4: Approach to the development of the safety case

FEP= Features, Events and Processes, PSAR= Preliminary Safety Analysis Report, FSAR = Final Safety Analysis Report



Indeed, it is considered likely that some of the current requirements and specifications will still change before the time of submission of the operations license application. Some of the modifications that do not affect the facility construction may be introduced in the Final Safety Assessment Report of the disposal facility. Otherwise, they will go through the change management system, which is part of the configuration management launched in early 2013 as a major activity area at Posiva. A potential change could be the adoption of the horizontal variant of the KBS-3 concept: this option is being studied as a possible alternative to the "classical" vertical KBS-3 concept, but at the moment it is still not possible to judge whether the horizontal alternative would offer safety or other benefits over the vertical option. In any case, the introduction of the new design would require an updated PSAR for approval by the regulator.

TURVA-2012 safety case

TURVA-2012 is the name of Posiva's safety case in support of the Preliminary Safety Analysis Report (PSAR, 2012) and application for a construction licence for a spent nuclear fuel repository at the Olkiluoto site according to the KBS-3 method. The license application was submitted to TEM (former KTM) on 28 December 2012. The Finnish nuclear safety authority, STUK, will review the safety case and related topical reports as part of its evaluation of construction licence application and the PSAR and give a statement on the construction licence application, which will form a basis for the government judgement on issuance of the construction licence.

The TURVA-2012 safety case presents the arguments for the long-term radiological safety of the planned disposal system. A plan for the safety case was first set up in 2005 and was later updated to better focus on quality assurance and control procedures and their documentation as well as on consistent handling of different types of uncertainties. Following Posiva's submission of the Interim Summary Report of the Safety Case 2009 (Posiva, 2010), STUK evaluated Posiva's preparedness to demonstrate long-term safety and operational safety and the fulfilment of the safety requirements for nuclear waste disposal against the governmental decree on nuclear energy (GD 736/2008). STUK's safety evaluation report (2011) provided feedback and advice that has been translated into key issues that are addressed within Posiva's RTD programme and in the development of the TURVA-2012 safety case. The feedback has also been taken into account in the systematic structuring of the safety case and the reports included in the portfolio shown in Figure 5. The portfolio includes:

- a description of the spent nuclear fuel to be disposed of in the geological repository;
- a description of the natural and engineered barrier system that the repository system provides, a definition of the safety functions and targets set for these, and a description of the present understanding of the processes that may affect the evolution and performance of the spent nuclear fuel, engineered barriers, host rock and the surface environment;
- a performance assessment systematically analysing the ability of the repository system to provide containment and isolation of the spent nuclear fuel for as long as it remains hazardous;
- a definition of the lines of evolution that may lead to failure of the canisters containing the spent nuclear fuel and to the releases of radionuclides (scenarios);
- analyses of the potential rates of release of radionuclides from the failed canisters, the retention, transport and distribution of radionuclides within the repository system and surface environment and the potential radiation doses to humans, plants and animals including the associated uncertainties and an evaluation of their impacts;
- the models and data used in the description of the evolution of the repository system and the development of the surface environment and for the analysis of activity releases and dose assessment;
- a range of qualitative evidence and arguments that complement and support the reliability of the results of the quantitative analyses;

- a comparison of the outcome of the analyses with safety requirements;
- a synthesis that brings together all the lines of arguments for safety, including the methodology, results and conclusions.

Aspects related to operational safety are dealt with in other parts of the PSAR.

Spent nuclear fuel must be kept isolated for as long as it could cause significant harm to the normal habitats for humans, animals and plants. In TURVA-2012 an assessment time frame of up to one million years into the future is considered.

Figure 5: The TURVA-2012 safety case portfolio

The portfolio consists of safety case reports (green boxes) and supporting reports (blue boxes); brief descriptions of the contents are given (white boxes). Disposal system = repository system + surface environment.

TURVA-2012

Synthesis

Description of the overall methodology of analysis, bringing together all the lines of arguments for safety, and the statement of confidence and the evaluation of compliance with long-term safety constraints

Site Description	Biosphere Description
Understanding of the present state and past evolution of the host rock	Understanding of the present state and evolution of the surface environment

Design Basis

Performance targets and target properties for the repository system

Production Lines

Design, production and initial state of the EBS and the underground openings

Description of the Disposal System

Summary of the initial state of the repository system and present state of the surface environment

Features, Events and Processes

General description of features, events and processes affecting the disposal system

Performance Assessment

Analysis of the performance of the repository system and evaluation of the fulfillment of performance targets and target properties

Formulation of Radionuclide Release Scenarios

Description of climate evolution and definition of release scenarios

Models and Data for the Repository System	Biosphere Data Basis
Models and data used in the performance assessment and in the analysis of the radionuclide release scenarios	Data used in the biosphere assessment and summary of models

Biosphere Assessment: Modelling reports

Description of the models and detailed modelling of surface environment

Assessment of Radionuclide Release Scenarios for the Repository System	Biosphere Assessment

Analysis of releases and calculation of doses and activity fluxes.

Complementary Considerations

Supporting evidence incl. natural and anthropogenic analogues

Main reports

Main supporting documents

Outcome and conclusions

The TURVA-2012 safety case demonstrates that Posiva's repository design and analyses of performance and safety are fully consistent with all legal and regulatory requirements related to long-term safety.

The performance of the repository system has been systematically analysed in different time windows. The analyses take account of the uncertainties in the initial state and in the subsequent thermal, hydraulic, mechanical and chemical evolution of the repository system, including the occurrence of unexpected or disruptive events. The analyses show that, under most conditions and lines of evolution of the host rock and engineered barriers, all performance requirements will be met. In this case, the copper canisters will remain intact and no releases of radionuclides will occur over at least one million years. Up to 50 000 years, the only plausible cause of release of radionuclides is that a canister with an initial penetrating defect escapes detection and is inadvertently emplaced in the repository. In the longer term, glacial episodes at the site may cause hydrogeological and hydrochemical changes leading to buffer erosion and increased canister corrosion, as well as seismic disturbances leading to shear movements on fractures intersecting the deposition holes. These changes and disturbances, if they were to occur, could potentially lead to the failure of up to a few dozens of canisters and to the release of radionuclides in less favourable locations within the repository.

Although releases of radionuclides to the environment are not expected, the safety analyses focus on the cases in which releases of radionuclides could occur. In the case of a canister with an initial penetrating defect, it is shown that, even if this happened to be emplaced in a location with relatively unfavourable local rock conditions, peak normalised radionuclide release rates to the surface environment are orders of magnitude below the radionuclide-specific constraints specified in the STUK YVL Guide D.5. In the long term (after approximately 100 000 years or more), the possibility of failure due to corrosion following buffer erosion and of failure due to rock shear need to be considered, but calculated radionuclide release rates remain below the regulatory constraint for the radioactive release to the environment, even for pessimistic and unlikely combinations of damage to canisters by rock shear events and erosion of buffer material due to dilute groundwater conditions.

Overall, it is concluded that the TURVA-2012 safety case demonstrates compliance with the legal and regulatory requirements for the planned and designed disposal facility for spent nuclear fuel at Olkiluoto. Uncertainties still remain in the data and models, and some of these are unlikely to be eliminated by further research. However, the analyses performed have shown that the repository system is robust with respect to these uncertainties and that the conclusions drawn about compliance with safety requirements hold true even when these uncertainties are taken into account.

The regulatory review of construction license application and the supporting safety case

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The Finnish disposal programme

Finland is one of the foremost countries in the world in developing the disposal of spent fuel. The construction license application for the Olkiluoto spent fuel disposal facility was submitted to the authorities at the end of 2012 and the facility is expected to start operation around 2020. This has been a long-term project with over 30 years of parallel development of the repository project and the regulatory approach to spent fuel management.

In 1983 the government made a strategic decision on the objectives and target time schedule for the research, development and technical planning of nuclear waste management. While an export and international disposal solution was still the preferred option, this decision required the licensees without this possibility to prepare for disposal in Finland and it also gave the time line for the milestones on the way to an operating disposal facility by 2020.

The licensing procedure for a disposal facility has several steps that are similar to all nuclear facilities in Finland and are defined in Nuclear Energy Act and Decree (1987, 1988). These licensing steps are:

- Decision-in-principle is required for a nuclear facility having considerable general significance. This is essentially a political decision: the government decides if the construction project is in line with the overall good of society. The decision can be applied for one or more sites, the host municipality has a veto right and the parliament has the choice of ratifying or not ratifying the decision.
- Construction license is granted by the government and authorises the construction of the disposal facility. The actual construction is regulated by STUK and includes several review and approval steps, hold points and viewpoints.
- Operational license is granted by the government and authorises the operation of the facility for a certain period. The operational license is needed before nuclear waste can be disposed.

The first step in the licensing process was reached at the end of 1999 when Posiva Ltd, the current implementer of the disposal programme, submitted the application for a decision-in-principle (DiP, 2000) for a spent fuel disposal facility in the Olkiluoto. The DiP was given by the government in late 2000, approved by the host municipality and ratified by the parliament in early 2001. It gave Posiva the authorisation to start to construct an underground rock characterisation facility, to the depth of the actual planned disposal, as required by regulation.

The safety case to support the decision-in-principle application was compiled for 1999. The safety case included a concept description, proposed site characteristics, general facility layout and a safety assessment. The application was further supported by an environmental impact assessment report.

STUK preparatory work for licence application review

The regulatory approach taken by STUK has been to have close follow-up of Posiva's safety case development and perform reviews of draft safety case documents. Another aspect has been to follow Posiva's R&D activities, which are described in RTD-plans, submitted to regulatory review every three years. In practice this has been implemented through regular visits to research laboratories, factories and workshops where safety related studies or demonstrations have been performed.

Posiva has submitted preliminary safety argumentation documentation for regulatory review since the decision-in-principle was ratified in 2001. This draft documentation has supported the development of the license application documentation. Hence, STUK has already been reviewing and assessing how the developing safety documentation meets regulatory safety requirements for 11 years. These preliminary findings have been communicated to Posiva and the target has been to identify and address the main safety-related concerns as early as possible.

Parts of the safety case was updated for authorisation of Onkalo construction and submitted to STUK in 2003. This documentation included:

- underground rock characterisation facility design requirements and layout;
- description of site baseline characteristics;
- assessment of construction disturbances;
- description of monitoring programme for construction period.

The Ministry of Employment and Economy required Posiva to submit a preliminary (draft) safety case by the end of 2009. The reasoning was to have a regulatory review of the status of construction licence application development. STUK reviewed the draft safety case and the process was used as an exercise for the actual licence application review. In STUK this was seen as a possibility to test the review process, review organisation and assessment of the preliminary safety case status.

The aim of the stepwise review, close follow-up and regular meetings with Posiva has been to identify the safety relevant issues and especially key safety concerns before Posiva finalises and submits the construction licence application. During the licence application preparatory phase STUK had a process for collecting and updating the position of key safety concerns with regular dialogue between STUK and Posiva. However after a while it was acknowledged that addressing single safety concerns did not in many cases lead to better overall understanding and sometimes the linkage to safety was not very clear.

Planning for the review and assessment of construction licence application

The review process, organisation, time schedule and resources are described in STUK's internal project plan for the licence application review. The main element of the project is of course the review of the extensive safety documentation. The assessment of safety requirement fulfilment and of implementer organisations' readiness for construction activities is supported with a STUK inspection programme for the pre-construction phase. The inspection programme is later broadened to encompass construction oversight of the encapsulation and disposal facility.

The regulatory assessment of safety is, of course, done according to regulatory safety requirements. As mentioned, the STUK approach was in earlier phases safety issue oriented and tended toward bottom-up assessment. To implement a more regulatory requirement oriented and safety-related review basis for detailed review and assessment, STUK started the development of a so-called review plan. This review plan is a collection of earlier regulatory observations and expectations for the construction licence application that are derived from and linked to regulatory safety requirements. The review plan is used as guidance for all experts participating in the review. It is also planned to be the structure for STUK's safety evaluation report.

The waste management and nuclear facility expertise that STUK has in-house has been allocated for the project. Parts of the safety case focusing on post-closure safety and on the actual safety assessment are extremely broad and need to be carefully assessed. Due to this reason STUK has agreements with VTT and several international experts for support in review and independent modelling.

Review of construction licence application and initial findings

Posiva submitted the construction license application and supporting documentation to the authorities at the end of 2012. STUK as a safety authority has started the review and assessment with a docketing review. The first phase is followed by thorough review and assessment against safety requirements and the outcome is documented in the authority's safety evaluation report. The planned time period for review and assessment is from 1.5 to 2 years. The time schedule is evaluated regularly to observe how the review process is proceeding.

According to the Nuclear Energy Degree, when applying for a construction license, the applicant shall submit the following to the Radiation and Nuclear Safety Authority (STUK):

- the preliminary safety analysis report, which shall include the general design and safety principles of the nuclear facility, a detailed description of the site and the nuclear facility, a description of the operation of the facility, a description of the behaviour of the facility during accidents, a detailed description of the effects that the operation of the facility has on the environment, and any other information considered necessary by the authorities;
- a probabilistic risk assessment of the design stage;
- a proposal for a classification document, which shows the classification of structures, systems and components important to the safety of the nuclear facility on the basis of their significance with respect to safety;
- a description of quality management during the construction of the nuclear facility, showing the systematic measures applied by the organisations that take part in the design and construction of the nuclear facility in their operations affecting quality;
- preliminary plans for the arrangements for security and emergencies;
- a plan for arranging the safeguard controls that are necessary to prevent the proliferation of nuclear weapons.

In addition to the documentation focusing mostly on operational safety, the regulations for nuclear waste disposal require the licensee to submit a safety case for post-closure safety. In practice, this is the lengthiest section of the construction license application documents. STUK YVL regulations give more details for the content of these documents.

During the first quarter of 2013 STUK performed first initial review phase. The aim of the initial phase, sometimes compared to docketing, was to verify that the licence application contains all the main elements requested in STUK YVL regulations. In other words, to check that coverage of the application is adequate for detailed safety review. The first STUK decision concentrated on coverage of operational safety documents, and based on the initial review the review progressed for most parts to the detailed review phase. However, some application documents were not accepted for review. The most important ones are related to safety classification, wherein the basis of safety relevance needs to be re-evaluated and also when needed the encapsulation system descriptions updated. The initial review for post-closure safety documentation is being finalised.

Oversight of future design and construction work

After passing the construction license step STUK will have comprehensive oversight for the detailed design, construction, fabrication and pre-operational testing, which will be followed by review and assessment of the operation license application.

STUK's regulatory activities (approvals, review and assessment, inspection) are implemented according to a graded approach. All the structures, systems and components (including underground rooms), of the facility are classified based on their significance to safety. The assessments of safety significance take into account both operational and post-closure safety. Since the management of the construction and related safety culture directly affect the safety and quality of the work and its long-term results, Posiva's management system is also subject to STUK's regulatory control.

STUK has developed a regulatory approach for oversight of the underground rock characterisation facility (ONKALO). This will form the basis for the development of the oversight system for disposal facility design and construction. The main difference is the more stringent requirements for selecting a suitable place for disposal tunnel location based on specific rock classification criteria.

For ONKALO oversight STUK has defined requirements for documents that are required to be submitted to STUK for review and approval. These documents include preliminary safety analysis reports, safety classification and descriptions of constructing quality assurance, design documentation, etc. In addition, Posiva was required to submit a plan on how it intends to communicate the progress of the construction work to STUK. The purpose of this document is to facilitate well planned, timely and properly targeted and resourced regulatory activities synchronised with the actual construction activities and provide timely information, for example, on unexpected events underground. These procedures will be broadened for the disposal facility construction phase. The main new element will be a more detailed rock suitability classification procedure and related regulatory oversight.

STUK's strategy has been to develop ONKALO oversight based on practices already implemented for other nuclear facilities. The development of regulatory activities has been an ongoing learning process. Based on the experiences of ONKALO oversight STUK is developing a regulatory framework for disposal facility oversight.

Summary

Posiva has submitted a construction license application for a spent fuel encapsulation and disposal facility at the end of 2012. STUK is currently finalising the decisions related to the initial review phase and starting the detailed review.

The preparatory phase has included among other things a systematic requirement and education programme, negotiation of framework agreements with external support experts and stepwise review of Posiva's developing safety case documents which included the draft construction license application documents.

STUK has started the development of oversight procedures for the construction phase. In this task, STUK can draw from experienced gained during ONKALO regulatory oversight.

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Session 3.7

Safety Case for Re-certifying and Operating a Facility: The United States Example

WIPP – Safety case evolution of an operating repository facility

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Introduction

The United States (US) Department of Energy/Carlsbad Field Office (DOE/CBFO) is responsible for managing all activities related to the disposal of transuranic (TRU) and TRU mixed waste at the Waste Isolation Pilot Plant (WIPP). The Environmental Protection Agency (EPA) is charged through the Land Withdrawal Act of 1992 to ensure DOE meets the disposal regulations promulgated in 40 CFR Parts 191 and 194 (EPA, 1993, 1998). The New Mexico Environment Department (NMED) further regulates WIPP to ensure the requirements of the Resource Conservation and Recovery Act (RCRA) are met through the Hazardous Waste Facility Permit (HWFP) for mixed wastes (State of New Mexico, 1999). The main function of the geologic repository disposal system is to limit radionuclide exposure to mankind and the environment. In the 1990s the DOE prepared the Compliance Certification Application (CCA) (DOE, 1996) document to provide the EPA with information to prove WIPP could meet the EPA's TRU waste disposal regulations in 40 CFR 191 and 194. This regulatory required documentation, along with additional scientific and engineering reports provided as references to the CCA, provided EPA enough data to determine WIPP could safely meet their requirements and the site was certified (licensed) in 1998. Since that time DOE has been re-certified twice (2006 and 2010) and made several operational changes to the waste disposal processes. The CCA documentation, the WIPP environmental impact documents, the re-certification documents and additional data, and the planned change information provided for EPA/NMED approval make up the present WIPP safety case.

1996 WIPP safety case

In 1996 DOE provided the CCA to the EPA to prove that WIPP could meet the disposal requirements the EPA had established and protect mankind and the environment from the effects of the radionuclides in the TRU waste slated for disposal at WIPP. This document included discussion of the geologic and engineered barriers to be utilised, the potential inventory to be disposed of, and the expected potential effects for any postulated future releases. Scientific journals and reports, engineered drawings, environmental impact assessments and statements, and performance assessment calculations were prepared, published, and included as part of the DOE argument that WIPP would operate safely and within the regulatory envelope. These documents, plans, procedures and drawings went through peer reviews and expert elicitations, were vetted in front of the public and both sets of regulators, and eventually culminated in a sufficiently convincingly set of data for EPA to grant WIPP a certification/license in 1998 and the NMED to grant a permit to operate in 1999.

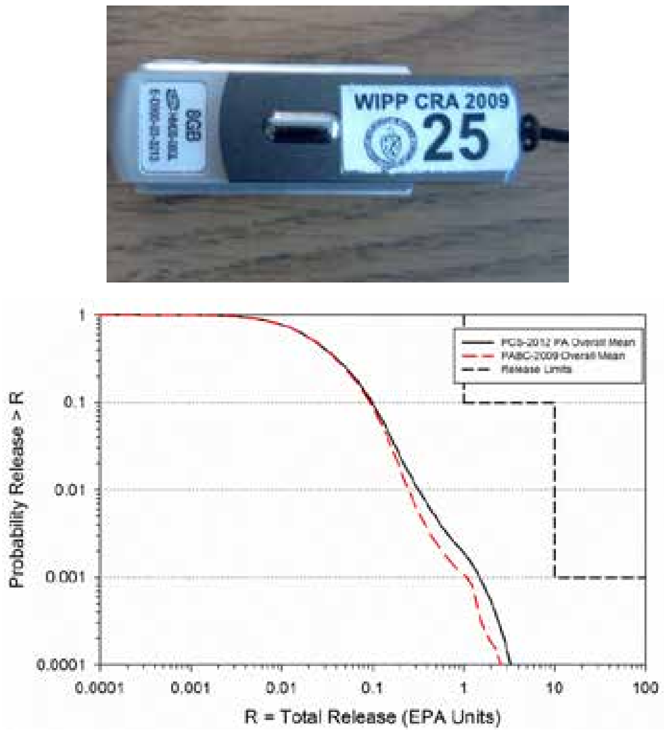
The 1996 WIPP safety case is an extensive collection of data, reports, descriptions, performance assessment calculations, environmental impact studies and assessments, and detailed engineering drawings and documents. The CCA itself consists of 9 chapters and 236 appendices, resulting in an excess of twenty thousand pages of data, that documents over twenty years of DOE research and development activities for the WIPP site in specific, and for salt repositories, in general. The information provided in the CCA, the WIPP EIS, and reports and analysis completed by oversight groups and in some cases the regulator, are all a portion of the WIPP complete safety case.

Re-certification safety case

The WIPP Land Withdrawal Act of 1992 requires the DOE to provide a compliance re-certification application (CRA) to the EPA every five years after the initiation of receipt of TRU waste. The first TRU waste was received at WIPP on 26 March 1999. The DOE submitted CRAs to the EPA in March of 2004 and 2009. The CRA is created to provide the EPA with documented evidence that WIPP continues to comply with the TRU waste disposal regulations and will continue to be protective of mankind and the environment.

Each CRA contains the data, reports and safety case information previously submitted to EPA and provides data on what has changed at the WIPP repository or in repository sciences in general that may have an impact on the safety case. The CRA focuses on the changes that have been made in the past five years, both major and minor, any new information from ongoing WIPP experiments and operations, and provides an updated performance assessment (PA) calculation to show compliance with the EPA set standard. The re-certification documentation, the most recent WIPP environmental analysis, and any additional reports and documents docketed by EPA or the NMED make up the WIPP safety case at each re-certification cycle.

Figure 1: WIPP CRA documentation is now placed on thumb drives and includes all information from past re-certifications, references and other sited documents, including a PA analysis



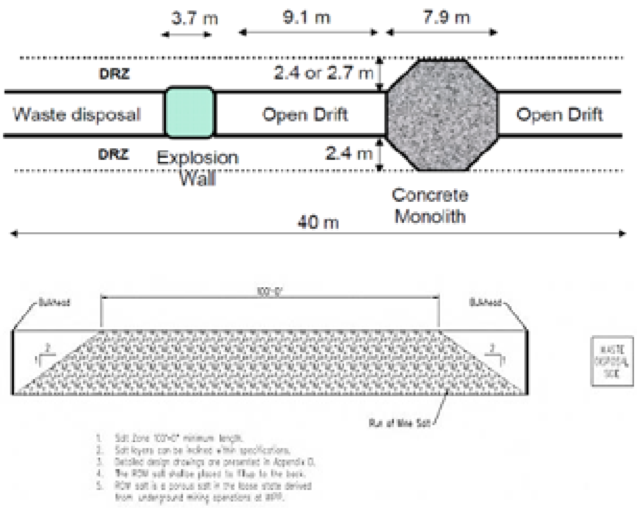
Operational changes and the safety case

As operations at WIPP have progressed over the past 14 years changes have been proposed, approved by the regulatory bodies, and implemented at the WIPP site. To make these regulatory changes, planned change requests (PCR) and/or planned permit modifications (PMR) are provided and discussed with either the EPA, or both EPA and NMED, the stakeholders and other interested bodies. PCR and PMR contain documents, reports, engineering drawings and performance assessment calculations to indicate the impacts, if any, of the proposed changes on operational and long-term safety. These approved planned changes in operations at WIPP are captured in the CRA's and become part of the safety case both during CRA and when they are approved for implementation. Two examples are shown below.

Figure 2: DOE submitted planned change requests (PCR) to EPA to reduce the amount of MgO emplaced to sequester the potential maximum amount of CO_2 generated. The EPA has approved these requests and now sacks filled to only 3 000 lbs. can be emplaced on top of alternating stacks of waste.

Figure 3: Panel closures (engineered barriers at the ingress and egress of each waste panel) are required by the EPA certification and the NMED permit for WIPP

The DOE has requested changes to the panel closure system design and schedule to both the EPA and NMED in a planned change request (PCR) and permit modification request (PMR) respectively



Each of these major changes, and others not highlighted here, can require numerous reports and documents that have implications for WIPP's protection of human health and the environment. As these changes are approved, the documentation is included in the regulatory record centres, and DOE's records and the documentation become a part of the overall repository safety case. These changes are also discussed, in less detail, in the re-certification following the change approval and the detailed documentation is referenced and becomes a part of that re-certification's safety discussion.

Conclusions

Nuclear waste repository operations are influenced by regulatory requirements. Documents developed to show how a repository will meet regulatory requirements, plus the environmental impact analysis and impact statements, along with additional data from stakeholders, oversight bodies and the scientific community at large, all can be docketed and considered as the safety case. Although this is not a concise, complete, one-document-contains-all type situation, such a document if it existed would be short-lived, as operational changes are made frequently during operations that change/ influence portions of the safety case. Five year re-certification cycles capture any changes large or small in a repository operation that may impact the safety case. At the WIPP repository these potential impacts to the safety case are also captured in PCR and PMR that are provided to the regulators and shared with the public, even more frequently.

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Continued oversight of the Waste Isolation Pilot Plant

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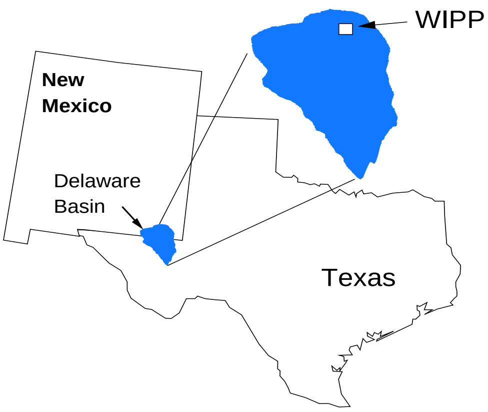
Introduction

The United States Environmental Protection Agency (EPA) developed environmental standards applicable to the disposal of defence-related transuranic wastes at the US Department of Energy's (DOE) Waste Isolation Pilot Plant (WIPP). By statute, EPA also serves as the regulator and implements these standards at WIPP, which has been in operation since 1999. The general environmental standards are set forth in the Agency's 40 Code of Federal Regulations (CFR), Part 191 *Environmental Radiation Protection Standards for the Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Wastes* (US NARA, 1985). These standards are implemented by site-specific compliance criteria at 40 CFR 194 (US NARA, 1996).

The repository waste area is ~650 meters below ground surface in a thick bedded salt formation that dips from west to east at ~1°. WIPP is located in the Chihuahuan Desert of south-eastern New Mexico, where the annual precipitation averages between 25 and 40 centimetres and there is high evapotranspiration (Figure 1). Much of the area around WIPP is federal land, managed by the Bureau of Land Management, and the area is sparsely populated.

Figure 1: Location of WIPP in the south-western United States



The transuranic waste disposed of at WIPP consists of materials such as radioactive sludges, soils and laboratory materials (e.g. chemical mixtures, contaminated glove boxes, paper and glass). Wastes are typically not treated unless necessary for shipping purposes (e.g. to limit hydrogen build-up). The waste is contaminated with plutonium, americium and other radionuclides, including some caesium and strontium. Transuranic waste is defined as waste with radionuclides heavier than uranium containing more than 3 700 Bq (100 nanocuries) of alpha-emitting transuranic isotopes per gram of waste; isotopes must have half-lives greater than 20 years.

The WIPP Land Withdrawal Act limits the total disposal volume to ~177 000 cubic meters (6.2 million cubic feet) and creates two categories of waste based on operational safety considerations. Contact-handled (CH) waste is defined by a container surface dose rate of less than 2 mSv/hr (200 mrem/hr) while remote-handled waste has a surface dose rate of greater than 2 mSv/hr. The contact-handled waste is disposed of in containers placed on the floor of the repository. Figure 2 shows some emplaced contact-handled waste. The statute limits the total activity of remote-handled (RH) waste to ~1.9×10^{17} Bq (5 100 000 Ci). RH waste is currently placed in holes bored into the waste room walls, but EPA has approved DOE plans to allow the RH waste to be placed in shielded containers (drums) that will reduce the dose rate to less than 2 mSv/hr, allowing the waste to be managed as CH waste. Although shielded containers are significantly heavier than the standard containers, this strategy allows DOE the option of placing some of the RH waste on the floor, supplementing the more limited volume that can be emplaced in the walls.

Figure 2: Contact-handled transuranic waste in the Waste Isolation Pilot Plant

Initial certification

After receiving DOE’s initial certification application in 1997, EPA approved the WIPP for operation in 1998 and DOE began shipping waste to WIPP in 1999. In the initial certification, EPA staff had the benefit of learning about the site and the technical issues for several years before entering into the formal certification decision-making process. During this period, EPA developed the site-specific compliance (implementing) criteria, and the agency was also able to incorporate into those criteria requirements that addressed topics that appeared to be weak in the DOE process, especially in the area of required documentation.

EPA applies a regulatory standard of “reasonable expectation”, which is a concept similar to the safety case approach. Reasonable expectation recognises the uncertainties and complexities in judging the performance of a disposal system over long time periods and encourages consideration of the full record in the regulatory agency’s decision process, including both quantitative and non-quantitative aspects (see e.g. 40 CFR 191.13). In its reviews, EPA looks to: *i)* understand the information provided by DOE, and determine whether there is appropriate and complete documentation (EPA’s Completeness Determination); *ii)* identify points of agreement and disagreement with the conclusions drawn by DOE; *iii)* determine whether the information provided by DOE is sound and provides a basis for a regulatory decision that can be defended technically and conforms to the regulatory requirements. In the review process EPA regularly communicates with DOE for clarification of existing information and may request new information as warranted.

For the initial certification, EPA relied upon DOE's modelling and reviews of the modelling to demonstrate the adequacy and quality of the modelling. The agency did not and does not conduct separate full performance assessments using separate codes, but has conducted and, as warranted, will conduct sub-system modelling as necessary (e.g. ground water or waste area specific modelling).

Re-certification and changes requested by DOE

The WIPP Land Withdrawal Act requires DOE to submit a re-certification application every five years after the initial receipt of waste. In the re-certification process DOE must identify changes that have occurred over the previous five years and analyse their impact on the potential long-term performance of the repository. Such changes could be related to the physical or chemical characteristics of the repository itself, or could arise from external factors, such as updates to the waste inventory. Once EPA determines that the re-certification application is complete, the agency has six months to review the application and make a final decision. During this review, EPA solicits and incorporates public comment. Since the agency went through an extensive review of and approval process for the modelling in the initial certification using external and contractor experts, verification modelling for re-certifications is primarily through modifying inputs to DOE's computer codes.

In addition, DOE must submit proposed changes to the WIPP repository to EPA for review and approval for those changes that could impact long-term performance. Since the WIPP is an operating facility, DOE periodically identifies potential changes that it would like to make for different reasons, such as disposal efficiency. For example, DOE requested, and EPA approved, a reduction in the amount of the required engineered barrier at WIPP. DOE conducted an analysis suggesting that it could save money while adequately maintaining the expected long-term conditions if they reduced the amount of magnesium oxide chemical barrier. The magnesium oxide is used to buffer the pH for control of actinide solubility and reacts with carbon dioxide if the waste panels are saturated with brine. EPA's initial review raised questions about DOE's analysis and required DOE to conduct a fuller study of the issue. EPA also conducted an independent analysis of the topic before agreeing to DOE's request. However EPA did impose a minimum limit on the amount of magnesium oxide that must be emplaced with the waste area and a requirement for DOE to track the carbon in the system to better ensure that there would be enough magnesium oxide to function as intended.

The planned change request may require the conduct of performance assessments outside of the re-certification process. The agency reviews the planned change request and re-certification information using an approach similar to that taken during the initial certification. Typically, these planned change requests are administrative (that is, not conducted through regulation) actions, though EPA has requested public comment on several of the requests.

The five-year re-certification requirement is advantageous in that it facilitates the continued communication between the agencies and makes it more likely that some key staff will overlap the review cycles, which allows issues to be resolved earlier and makes the process run more smoothly. This is illustrated by the difference between the first and second re-certification applications. EPA's review of the first application, submitted in March 2004, identified a number of areas where more information was needed, so that the application was not deemed complete for nearly two years. By contrast, with a much better idea of the needs of the regulator, DOE's 2009 application needed only limited additional information. A disadvantage of the five-year cycle is that it can seem as though one application has just been submitted when work on the next must begin. A longer cycle might allow more “breathing room” and opportunity to address other issues, but would also make it harder to retain the institutional knowledge necessary to conduct the review efficiently.

Observations about the process from initial approval through operations

As stated previously, the initial certification presented a number of challenges to both the EPA and the DOE. Both agencies responded to this precedent-setting situation with high-quality technical and managerial staff. In addition, the National Academy of Sciences had been involved with the programme in an ongoing independent review role and the DOE WIPP programme benefitted greatly from those interactions, as did the EPA. Nevertheless, the issues that needed to be addressed technically and from a regulatory perspective were numerous. The EPA had to identify the most important issues and prioritise its resources to those issues. The agency required the DOE and itself to produce high quality documentation that addressed the technical issues and the regulatory requirements in a thorough and defensible manner.

For the re-certification process, the approach is similar, but there are fewer issues that need to be addressed from first principles. Because of the requirement to account for items that have changed, the site developer (DOE) needs to be cognisant of the scientific literature and advances in knowledge for topics that could affect performance. The regulator also needs to maintain an awareness of the issues. For example, after WIPP was certified, new information became available about microbial extremophiles that needed to be considered for their potential impact on WIPP. Actinide solubility research continues to become available in the literature, being generated by the WIPP programme itself or outside researchers.

At some point in the application process there will be a need to transition from a focus on primarily site characterisation and research to translating the research information into compliance demonstration. Different personnel may be needed because some of the researchers may not be able transition from research to compliance. Nevertheless, the developer needs to be able to maintain the requisite expertise to address technical issues that are certain to come up.

Maintaining expertise will be an issue even after the facility is operating and is a challenge for both the implementer and regulator. This is because site developers and regulators need to have mechanisms to address changes, both large and small, after the facility begins operation for reasons as varied as efficiency, cost, safety and regulatory compliance. A process or procedure needs to be developed and agreed to by all entities on how and when changes will be made. To some extent, the five-year re-certification requirement constrains the change process, as both agencies know there will be periods where the DOE is focused on preparing the application and the EPA is focused on reviewing it, so that other potential changes will have to be strategically pursued to avoid losing momentum.

Maintaining expertise over longer times is a continuing issue because regulator and developer staff will move on to other projects and retirements. Some EPA staff have retired or taken new jobs, and while fortunately the core contractor staff have remained present, they too will be unavailable at some point. The DOE scientific staff have had a great deal of turnover since the initial certification. One thing that EPA staff did to help document the process after the initial certification was to develop a document and a series of slides to capture key lessons from the experience. The information from that effort is still being used as the third re-certification approaches.

Summary

In summary, the initial certification of the WIPP required an extensive effort. The five-year re-certification process is similar but is less intense with fewer issues, and the site developer still has to develop high quality information. The site developer can focus on those areas that have changed, and there will be further changes during the operational period of the facility. One advantage of the re-certifications is that both the site developer and regulator can build upon the existing knowledge base instead of dealing with everything as a new issue; however, staff turnover brings challenges to maintaining the knowledge base.

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Session 4

Staff Management, Training and Knowledge Management

Staff management, training and knowledge management

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Staff management/training and knowledge management are organisational issues that are particularly sensitive in long-term projects stretching over decades like the development and operation of a geological repository. The IAEA has already issued several publications that deal with this issue (IAEA, 2006, 2008). Organisational aspects were also discussed in the framework of a topical session organised by the Integration Group for the Safety Case (IGSC) at its annual meeting in 2009 and were regarded as a topic deserving future attention (NEA, 2009a). More recently, the Forum on Stakeholder Confidence (FSC) identified organisational, mission and behavioural features as attributes of confidence and trust (NEA, 2013). They also identified that aspects such as structural learning capacity, high levels of skill and competence in relevant areas, specific management plan, good operating records, transparency and consistency are associated with confidence building in a safety case. These aspects are considerably related to staff training/management and knowledge management.

The IGSC has initiated a proposal of study dedicated to staff training/management and knowledge management with the objective to highlight how these recent concerns and the requirements issued by the IAEA are concretely implemented in the national programmes. The goal of this study is to acknowledge the differences of views and needs for staff management and knowledge management at different stages of individual programmes and between implementer and regulator. As a starting point to this study, the JAEA and ONDRAF/NIRAS prepared a draft questionnaire in order to succinctly capture processes and tools that the national organisations have implemented to meet the requirements and address the issues set out in the field of staff and knowledge management.

For the purpose of this study, a questionnaire is now under development, which will be presented on the occasion of this symposium with guidance based on a test run organised to evaluate the efficiency of the questions posed, prior to its sending to sister organisations.

The draft questionnaire intends to cover the following four items related to staff and knowledge management:

- The first question addresses the resources dedicated to a development programme of geological disposal and their allocations among the different activities. These evolve over time, from one development stage to the next but also due to societal evolution. As an example, the NEA (2007) noted that the growing involvement of the stakeholders in the waste management programmes from the end of the

20th century produced a fundamental change in the cultural and structural aspects of the waste management organisations. Specific resources were indeed deployed to strengthen stakeholder dialogues as well as external and internal communications. An objective in this survey is to analyse more globally how the resources of a national programme have evolved since its inception towards its implementation and how specific processes (e.g. QA processes), competence (e.g. special experts, safety assessors, co-ordinators, etc.) or stages (e.g. generic, sitting and operational phases) impacted the financial and human resources. Input from the respondents should allow identifying how the national organisations adapt to the increasing diversity and needs of the RD&D activities as a geological programme moves forward and also how they provide, maintain and plan tailored competence to fulfil these needs.

- The second topic touches on the traceability of the reporting: programmes, in particular its internal reporting and the accessibility to the information and related decisions. The issue focused on here reflects the discussion that arose at the 2009 topical session of the IGSC concerning the structuring of the information, for example, its transparency and categorisation as safety-critical or, at the opposite, not safety-relevant, or even obsolete.

- The third topic is related to the application of QA/KM measures. Implementation of comprehensive QA/KM over data/information/knowledge, computer codes and documents is highly challenging and very elaborate and complex process. A graded approach is necessary to maintain the focus on the project objectives. The questions for this third topic are directed toward how such approaches are implemented or reflected in the regulations.

- The introduction of the safety functions in geological disposal programmes emerged about a decade ago from the multi-barrier principle. Safety functions embody key aspects of long-term performance of the geological disposal system after closure, which can then be developed and translated into a hierarchical structure of technical and functional requirements (NEA, 2008). The bridge that safety functions provide between technical and scientific knowledge on the one hand, and between safety and feasibility objectives on the other, is a valuable strategic tool used by waste management organisations to steer and structure their R&D programmes (NEA, 2009b). The usefulness of a requirement management system has also been recognised by the NEA (2006) in the framework of a workshop on the design of engineered barrier systems (EBS). As the disposal programme moves forward, other dimensions, such as the operational safety, the feasibility, the costs and the environmental impacts, emerge and combine with long-term safety. The requirements and concerns expressed by the stakeholders representing different groups should also be accounted for and would, in some cases, conflict with technical requirements. The measures for this requirement management issue are, for examples, the practical tools developed by:

  – the national implementers to structure this diversity of requirements in a synoptic and hierarchical way;

  – the regulatory bodies to ensure the development of a structured and comprehensive set of regulations and guidances and to analyse whether the implementer’s safety case meets the regulatory requirements.

The questionnaire and test run results will be further discussed at the symposium to explore the possibility of international co-operation in this area by e.g. sending the questionnaire to interested sister organisations.

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Session 5

Keynote Lecture on Long-Term Governance of CO_2 Storage

Specific issues and long-term governance of CO_2 storage

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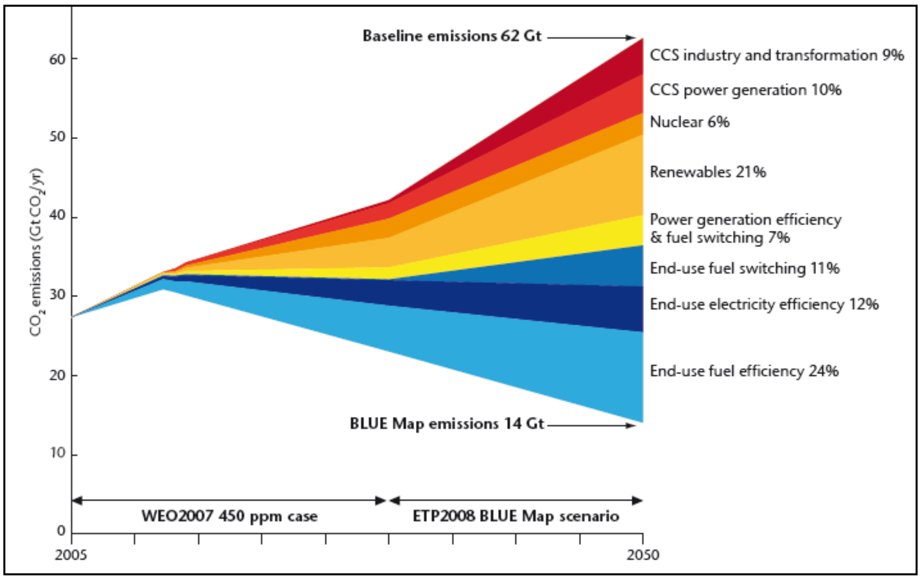
CO_2 capture and storage (CCS) is one if the solutions that is presently envisaged, together with other solutions, in order to limit the anthropogenic emission of CO_2 – CO_2 being one of the most important greenhouse gases. It is a full chain that consists in capturing the CO_2 at the emitting plant (e.g. a power plant that consumes coal or gas), then transporting it to a suitable location and injecting it in deep underground layers for definitive storage. After presenting the main features of the technology, this paper will describe its main issues and challenges at the present state, in order to show the similarities and differences between CO_2 storage and nuclear waste repositories. The final part will focus on the social issue and needs in terms of governance.

Figure 1: CCS, a 19% contribution to a global strategy to limit CO_2 emissions (IEA, 2008)



The primary issue: Feasibility from the technical and economical perspectives

The first CCS storage has been operated since 1996 at Sleipner (North Sea, Norway). In order to meet its objectives CCS should have more than 1 000 plants world wide by the year 2020. Despite the efforts of operators and the help of public funding, less than 20 full scale projects are operating today; this shows the overall challenge for this technology.

The first difficulty is to target a sufficient underground storage capacity, close enough to surface plants where CO_2 is to be produced and captured. Another strategic issue is of course to reduce the cost, which lies mainly in the capture process. At present, the cost for 1 tonne of CO_2 captured and stored is more than five times higher than the cost (or penalty) for emitting it.

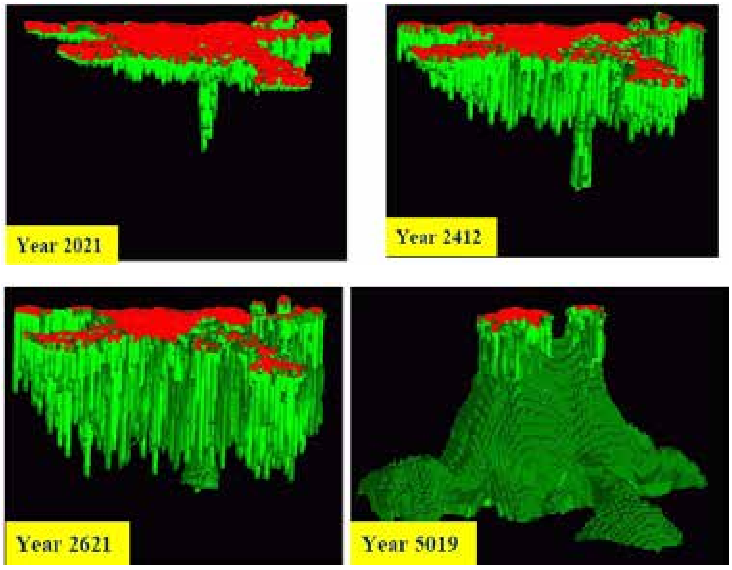
A wide issue: Long-term efficiency and safety

It is essential to ensure both the long-term efficiency of storage (no re-emission of the CO_2 stored) and its long-term safety (no harmful effects to the environment). Both issues imply no leakage.

From the methodological point of view, we can identify similarities with nuclear waste repositories: the need to envisage long-term evolution, the need to study many different underground phenomena, the need to envisage all possible scenarios – including scenarios of "normal evolution" and scenarios of "altered evolution". The full paper presents the main scenarios at stake.

As concerns storage, as for radioactive waste, we are faced with uncertainties and our present non-perfect knowledge of what is likely to happen over the longer term. Therefore monitoring, on the one hand, and modelling, on the other hand, play key roles.

Figure 2: A prediction of CO_2 fate in the longer term at Sleipner (Torp, 2007)



Last but not least, a final issue: The need for *ad hoc* governance at several levels

The CCS technology in itself aims at a global, world-wide efficiency. The full paper shows that has it benefits from the support of several bodies at a global level. A recent initiative is the standardisation effort that was engaged in 2012 though ISO Technical Committee TC-265. Yet, the permits depend from national and/or local procedures and any negative consequence would have to be managed at a local level. Hence risk prevention and social awareness also require local action.

Some recent examples showed that a good technical demonstration is not sufficient to ensure social acceptance and/or public engagement, especially in Europe. Transparent communication towards regulators and towards the public is necessary and, more than that, a debate with different stakeholders. As shown before, uncertainty has to be managed, but not only from a technical point of view. Because of social involvement in the longer term, the role of experts may also be questioned.

All actors implied in CCS development should exchange knowledge about what works and what does not work. This strategy of “learning from experience” probably goes beyond scientific conferences and may be a help to manage the remaining uncertainties. If the CCS technology does not develop rapidly, it will not meet its objectives to reduce greenhouse effect.

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Session 6

Specific Issues and Challenges in Safety Case Development

Session 6.1

Specific Issues and Challenges in Safety Case Development, Part I

TURVA-2012: Handling QA

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Goals and principles

Posiva applies a management system that complies with the ISO 9001:2008 standard for all activities, including the production of safety case reports, and requires the application of the same quality assurance principles from all its contractors and suppliers. The ISO standard was first launched in 1997 and has since been subject to continuous maintenance, updating and several internal and external audits.

The purpose of Posiva's quality management system is to ensure, in a documented and traceable way, that Posiva's products – whether in the form of abstract knowledge and information, published reports or physical objects – fulfil the requirements set for them. The general quality objectives, requirements and instructions defined in Posiva's management system also form the foundation for the quality management of safety case activities.

The quality management of the safety case follows the Posiva's general management system, which is based on the ISO 9001:2008 standard and management through processes, but also applies the principle of a graded approach similar to the safety guides for nuclear facilities. This means that the primary emphasis in the quality control and assurance of safety case activities is placed on those activities that have a direct bearing on the arguments and conclusions on the long-term safety of disposal, whereas standard quality measures are applied in the supporting work. The quality management of the safety case aims at traceability and transparency of the key data, assumptions, modelling and calculations.

Regarding the activities related to ONKALO, the management system also takes into account the regulatory requirements of YVL Guide 1.4 "Management System for Nuclear Facilities" (which will be subject to revision in 2013).

Application to TURVA-2012 safety case production

The overall plan, main goals and constraints for the TURVA-2012 safety case production process are presented in the Safety Case Plan (Posiva, 2008). The details of how the Safety

Case Plan 2008 was implemented are described in the long-term safety project plan. The work is managed and co-ordinated by a core group and supervised by a steering group.

A specific quality co-ordinator (QC) has been designated for the activities related to the quality assurance measures applied to the production of the safety case contents. The QC is responsible for checking that the instructions and guidelines are followed and improvements are made in the process as deemed necessary. The QC is also responsible for co-ordination of the external expert reviews, maintenance of schedules, review and approval of products, and management of the expert elicitation process. The QC also leads the quality review of models and data used in the Data Handling and Modelling sub-process (see below). Regular auditing of the safety case production process is carried out as part of Posiva’s internal audit programme.

The production of the safety case is divided into four main sub-processes: Conceptualisation and Methodology, Data Handling and Modelling, Safety Assessment, and Evaluation of Compliance and Confidence (Figure 1).

- The Conceptualisation and Methodology sub-process frames the safety case, providing the description of the disposal system, features, events and processes (FEP) analysis and the formulation of scenarios, including system evolution. It guides the definition of the performance targets for the EBS and the bedrock, which form the core of the requirements management system (VAHA), and the basis for the development of an approach to evaluating the suitability of the rock at various scales through application of the rock suitability classification (RSC) system.

- The Data Handling and Modelling sub-process makes the connection between Posiva’s main technical and scientific activities and the production of the safety case. It assembles the models and key input data used in the safety case for describing evolution, radionuclide transport and dose assessment. It also identifies key data needs to be developed. It addresses models and data related to the spent nuclear fuel, the EBS and the site (including both geosphere and biosphere).

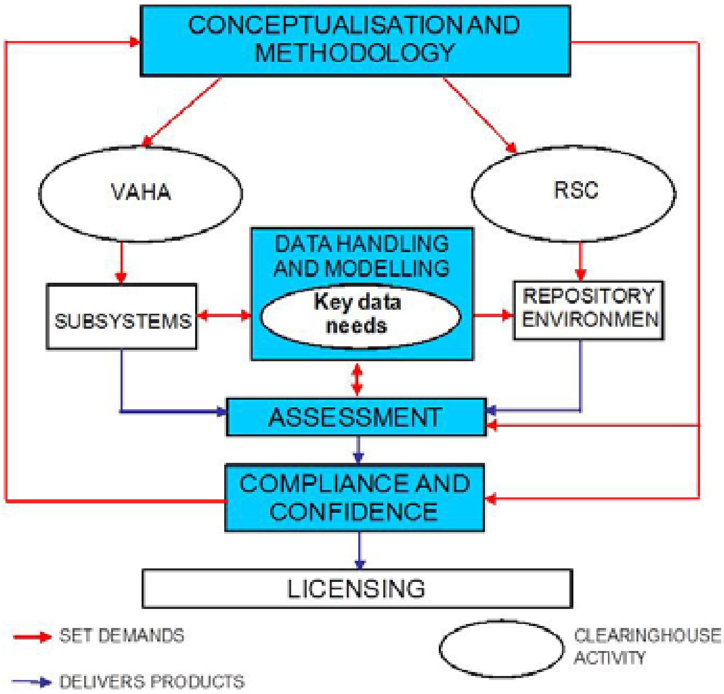
- The Safety Assessment sub-process identifies the lines of evolution that could lead to the release of radionuclides and formulates the scenarios that are analysed first to quantify the releases from the repository system to the surface environment and then to quantify the radiological impact of those releases to humans, plants and animals.

- The Evaluation of Compliance and Confidence sub-process evaluates compliance of the assessment results with the regulatory criteria and the overall confidence in the safety case, taking into account the completeness of the scenarios considered, uncertainties within the assessment and complementary considerations regarding the long-term safety of geological disposal.

It is essential that the information and data passed between sub-processes be quality assured. The main safety case portfolio reports *Models and Data for the Repository System* and *Biosphere Data Basis* and *Biosphere Assessment* act as the quality assured interface between the safety case activities and the research and technical activities: they include all the essential EBS and site information and data needed for the performance and safety assessment calculations, while more details can be found in the supporting background reports, such as *Site Description*, *Biosphere Description* and various *Production Lines reports*. The quality of *Site Description* is mainly ensured through the application of scientific principles, while the methods of quality control for the design and implementation depend on the nature of the materials and technology in question.

Figure 1: Main activities of the safety case (production) process

VAHA – requirements management system, RSC – rock suitability classification system



Model qualification and code verification

A range of quality control and assurance measures has been adopted to promote confidence in the models (and codes) and hence to promote confidence in the analysis of the calculation cases. According to Posiva's Safety Case Plan (Posiva, 2008), these quality control and assurance measures comprise:

1. validation of input data for the scenarios and models considered; the limits of applicability of the input data are checked against the assumptions related to the scenarios and models;
2. validation of the models used to analyse the scenarios;
3. verification of assessment codes;
4. validation of the assessment codes for the intended application;
5. documentation of input for the production runs;
6. application of a procedure to ensure codes are correctly applied;
7. documentation of the code versions used;
8. reporting of non-conformities.

Measures 1 and 2 relate to the quality of models and to the selection and checking of data that are implemented in the codes. Actions undertaken to validate and thus promote confidence in the models and data used in TURVA-2012 are described in the reports *Models and Data for the Repository System* and for the surface environment in *Terrain and Ecosystems Development Modelling*, *Surface and Near-Surface Hydrological Modelling*, *Biosphere Radionuclide Transport and Dose Assessment* and *Dose Assessment for Plants and Animals*.

At a more general level, the reports *Features, Events and Processes* and *Complementary Considerations* summarise the understanding of processes relevant to repository performance and safety that can be gained from observations at the site, including its regional geological environment, and from natural and anthropogenic analogues for the repository and its components.

Measures 3 to 8 relate mainly to the selection, testing and application of computer codes used for the radionuclide release and transport calculations and for dose assessment and to the documentation of results. Actions undertaken to verify and thus promote confidence in the computer codes and their application are described in *Assessment of Radionuclide Release Scenarios for the Repository System* and *Biosphere Assessment*.

Verification measures, including benchmarking exercises that address specific functions of the main computer codes used, have been carried out during their development. In addition, benchmarking exercises have been carried out in which results generated by these codes were compared with those generated by other well-tested codes that have been shown to handle the main features, events and processes of relevance. The benchmarking exercises where possible use test cases that are representative of the types of calculations needed for TURVA-2012, and so contribute to validation as well as verification. Finally, external reviews of newly developed codes have been carried out and deficiencies identified in the reviews were addressed before the calculations for TURVA-2012 were undertaken. Based on all these measures, it is concluded that the main codes have been verified and validated to the extent required for use in TURVA-2012. Regarding code application, numerical parameters, such as the size of time steps, are carefully selected to ensure that the model results are sufficiently accurate. A version management system has been used to keep track of any changes in input files and thus maintain the reproducibility of calculation results. An assessment database has been set up for the storage, checking and exchange of input data, intermediate results and final results. Finally, an electronic system, termed *docgen*[1], has been developed to keep track of, and to archive, the results of safety assessment calculations as they are produced. Results of model calculations and their associated input files are downloaded to *docgen* automatically from the assessment database. In this way, it has been possible to follow the progress of the calculations and carry out quality assurance and plausibility checks in a timely manner.

Data clearance

A wide variety of data have been used for the compilation of the safety case. An important activity for ensuring the quality, transparency and consistency of the data used in the safety case is data clearance. Data clearance is the formal procedure to approve the data to be used as input to the models used in the analyses reported in the safety case, such as the assessment of the performance of the repository system, the analysis of the release scenarios and the analysis of radiological consequences.

The raw data produced, e.g. by site investigations or laboratory tests, are usually not directly suitable for the models used in the safety case, and further interpretation and modelling are needed. Sometimes there are no site-specific data available, thus literature data and data from other sources, e.g. data produced for other nuclear waste management organisations, need to be used. The applicability of the data for the specific purpose and

1. The docgen system was originally developed for Nagra, the Swiss National Co-operative for the Disposal of Radioactive Waste. The version used in TURVA-2012 has been extended and adapted for Posiva.

conditions analysed in the safety case is assessed as part of the data clearance process, and potential sensitivity cases to be addressed by modelling are suggested. The data may be in the form of single parameter values, a range of parameter values or a probability distribution function. In some cases, different data apply to specific model variants or versions (e.g. applying to a specific hydrogeological model or repository layout).

The data clearance procedure consists of the following steps: *i)* identification of the data needs; *ii)* collection of suitable data; *iii)* documentation of the suggested data, their intended use and justification for their selection; *iv)* data approval. Separate reports on various categories of data collection have been produced, e.g. regarding climate evolution, solubility and sorption data for the near field and far field, and earthquake frequency.

The clearance process has been implemented according to guidelines that address the documentation of data sources and quality aspects. Single items of data and databases are approved through a clearance procedure supervised by the long-term safety Quality Co-ordinator. Process owners check and approve the data while the Quality Co-ordinator checks and approves the procedure. Data used may also be approved using other Posiva databases like the requirements database VAHA or investigation database POTTI and the respective approval processes. A clearance procedure has been applied to all key data used in the performance assessment (i.e. showing compliance with performance targets and target properties), and in safety assessment (i.e. radionuclide transport analysis and dose calculations).

Where data are subject to particularly high levels of uncertainty, further review of the data by subject matter experts and safety analysts has been carried out and a formal expert elicitation has been applied. Purpose-specific databases have been applied to manage the data clearance procedure in a structured way and to ensure the controlled use and traceability of input data used as input to safety related assessment calculations.

The expert elicitation process has been applied to a specific case (solubility and sorption data) to identify the main sources of uncertainty and determine whether different views may have to be propagated through the safety assessment. This expert elicitation process was initiated, carried out, documented and managed by the long-term safety Quality Co-ordinator. He also recruited the independent experts involved in the process. The clearance procedure is documented in *Models and Data for the Repository System*, *Biosphere Data Basis* and *Biosphere Assessment*.

The *Models and Data for the Repository System* and *Biosphere Data Basis* give an overview of the modelling carried out within the safety case and how the different models link to each other. They also present the key models and data used in the safety case. For each model, the conceptual model, the mathematical model and the codes used are described. This description covers the key assumptions and simplifications. The uncertainties related to modelling and their impact on the results is presented. Also, possible alternative models are discussed and the basis for selection of the specific model is given. Discussion of the data describes the production, qualification and uncertainties related to the data as well as potential alternative data.

In order to assess confidence in the models and data, the applicability of the models and data to the conditions at Olkiluoto and to the safety case purposes as well as the applied quality measures are discussed. Further, the impact of the uncertainties in the models and data on the modelling outcome is assessed and, where necessary, needs for model and data improvements are identified.

Report and product review and approval process

The review and approval process of the safety case production (i.e. main portfolio reports) has been carried out in a fully traceable manner. This has included, first, an internal review by safety case experts and subject matter experts within Posiva's research and development programme and, second, a review by external experts. A group of external experts covering the essential areas of knowledge and expertise needed in safety case production has been set up. The review comments are managed via review templates, which record the review comments and how each comment has been addressed. Upon completion, this template is checked and approved according to the quality guidelines of Posiva.

Quality assurance and quality control measures related to the production and operation of the repository are discussed in detail in the Production Line reports (*Canister, Buffer, Backfill, Closure* and *Underground Openings Production Line*).

Further development

The quality management system is being further developed along with the future steps in the RTD programme when going further towards the construction phase. Important aspects considered in the development of the quality management system are also the regular feedback given by the authorities in their auditing of Posiva's quality management system.

The role of quality management in safety case development – Nagra's experience

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Introduction

This paper discusses the role of quality management (QM) in safety case development based on Nagra's experience from a broad range of projects. These include Project Gewähr (L/ILW & HLW, Nagra, 1985), the Wellenberg Project (L/ILW, Nagra, 1994), Project Opalinus Clay (HLW, Nagra, 2002a, 2002b), and recent project work needed in the context of the Swiss site selection process (L/ILW & HLW, Nagra, 2008a, 2008b, 2008c, 2010).

Broadly speaking, Nagra's Quality Management policy is focused on ensuring: *i)* the quality of the disposal system (siting, design and implementation); *ii)* the quality of the underlying scientific understanding, which are seen as key elements of a credible safety case, along with the quality of the safety calculations themselves and of compiling the safety case, including the drawing of conclusions (Nagra, 2002a). All aspects of QM discussed in this paper should be seen in this context.

Importance of QM

The most important principle to ensure sufficient quality (i.e. no unacceptable deviations from the planned scope of work) is a company policy that actively promotes a quality culture and awareness throughout the company. This begins by recruiting personnel that: *i)* are sufficiently qualified for their work and have the necessary tools; *ii)* are fully aware of the expectations placed upon them; *iii)* take full responsibility for their work. The requirement of sufficiently qualified personnel applies both to the Nagra staff involved as well as to personnel of contractors. Central aspects of the quality culture are openness to new developments in science and technology and the handling of detected errors.

To promote a quality culture among staff, it needs to be recognised at all levels in the company that errors are inevitable, no matter how comprehensive the company QM system, and that this is acceptable provided QM procedures have been followed with reasonable diligence. The important point is that errors, once detected, are openly acknowledged and appropriately handled. This will generally involve: *i)* an assessment, involving all relevant staff and contractors of the importance of the error for the current project; *ii)* the implementation of corrective actions if the importance of the error so demands; *iii)* the definition and implementation of measures to help avoid similar errors in the future. Nevertheless, the main emphasis of the quality culture and the QM procedures is on avoiding errors. The corresponding key elements are discussed in the next section.

Key elements of Nagra's QM system

The key elements of Nagra's QM system relevant for technical project work are schematically shown in Figure 1, and discussed below (see also Nagra, 2002b).

Figure 1: Key elements of Nagra's QM system relevant for technical project work

- *Project Plan* – Through a project plan developed at the start of the project, all persons involved in the project are fully aware of the aims and boundary conditions of their work (definition and analysis of project mission and identification of sub-projects), are clear about the responsibility they have within the project (organisation of project), and are aware of the inter-relations between the different activities/sub-projects within the project, the timing of these activities and the expected deliverables. The project plan also defines the need for project plans for sub-projects and identifies for which activities/(sub-)projects a quality management plan is needed. If needed, the project plan is updated.
- *Quality Management Plan* – With the quality management plan developed early in the project, it is clearly defined what quality assurance (QA) measures are required for which activity/(sub-)project by whom and at what stage in the project (*a priori* definition) and the QA measures conducted are clearly documented (*a posteriori* documentation). The necessity for a (*a priori*) quality management plan for (sub-)projects is defined in the project plan. The QA measures identified in the quality management plan can either be defined by specific quality management guidelines or can specifically be defined. If needed, the quality management plan is updated. An integral part of the QA measures is the requirement of a peer review for all Nagra technical reports.
- *Data Clearance* – A rigorous data clearance process ensures that throughout the whole project suitable and consistent data are used. Data are always cleared for a certain purpose and for specific (sub-)projects/users. It is the duty of a data user to initiate the clearance process for a specific purpose and to ask the data producer to deliver the data in a suitable document, together with a data clearance form, to the clearance group for processing. The clearance group ensures (by expert judgement)

that the data are adequate for the intended purpose. If the data are especially important, additional QM measures (e.g. peer review) are implemented in the process of compiling/producing the data to be cleared. Whenever any planned updates, new findings or errors detected in the course of the project need to be considered, a revised data clearance form and supporting document is issued and submitted again to the clearance group for processing. The data clearance process ensures that all data users are made aware of such a revision. This forms the basis for further actions, e.g. a re-calculation of the affected assessment cases. Such further actions are, however, not part of the data clearance process. If required, they are initiated by the project manager in charge.

- *Project Documentation* – Thanks to a project documentation that is continuously updated, all persons involved are fully aware of the relevant documents for the project. For this purpose a documentation structure is defined, a centralised catalogue of all relevant documents is maintained and all those documents catalogued that are not readily available (e.g. Nagra internal notes) are archived. At the end of the project all documents entered into the project documentation are checked for their relevance to ensure that the documents needed for traceability of the project are properly catalogued and available for any future use.
- *Audits* – To ensure the effectiveness of the QM system, audits are performed to check if the procedures are followed and if there is a need to modify or enhance the existing QM system. If any deficiencies are detected, appropriate measures are taken by the project manager in charge. The audits may also lead to modifications and refinements of the QM system.

These principles and measures described thus far are common to all significant projects undertaken by the company, and do not relate exclusively to safety case development. However, for individual working steps within a project, specific working procedures are defined that may include specific QA measures, defined in specific QM guidelines, which are specific to the type of project at hand. For example, for performance assessment calculations carried out within any significant project, specific QM guidelines address the issues of: *i)* performing safety calculations; *ii)* checking safety calculations; *iii)* maintenance of computer programs for safety calculations. These QM guidelines have more recently been updated and extended. As an example, the current guidance regarding the use of computer codes for safety calculations is summarised in Box 1.

Box 1: Summary of current QM guideline regarding the use of computer codes for safety calculations

- The adequacy of the safety assessment codes for the different calculation cases and the adopted modelling approaches need to be checked (explicit/implicit consideration of processes and parameters, acceptable simplifications) and the codes need to be verified to the extent required by the modelled problem.
- Attention has to be given to the systematic naming and structuring of the assessment cases and the corresponding input and output files.
- Input data need to be traced to, and checked against, their source, and, if relevant, to the corresponding data clearance.
- Version control has to be implemented for all program, script and input files, and log files with specific information on the execution of each calculation case must be created.
- QA measures, such as consistency checks of input and calculated data, must be carried out and may be integrated into the log file. Automatic control measures both at code execution and post-processing of results have to be carried out.
- QA measures must be documented in technical notes and added to the QA documentation of the project.

Guidelines also relate to supporting activities for safety calculations or safety case development. For example, quality guidance is given on the acquisition and development of safety analysis codes and on the maintenance of these codes. This guidance aims to ensure, among other things, the operational readiness of the compiled version of a code and its computational environment (including pre- and post-processing) for safety analysis when required.

Regarding waste acceptance criteria, which also support the safety case, Nagra has developed and implemented a set of rules regarding waste documentation and acceptance; formal disposability assessments have been performed since 1989 (Zuidema, *et al.*, 1997).

Description and documentation of QA measures – example Project Opalinus Clay

In the Project Opalinus Clay documentation, the general QA measures were described in a Nagra report (2002b). Specific planned QA measures were set out in an *a priori* quality management plan and documented in an internal note. The QA measures actually taken during the project were collected in specifically designed QA forms, signed by the responsible project managers and documented after completion of the project in an internal note. Safety calculations and the corresponding preparatory work, including the development of models, codes and data, were conducted within the framework of the Nagra quality management system and took into account the requirements defined by the specific QM guidelines discussed above. Specific QA measures taken during the project regarding the use of computer codes for safety calculations were recorded and documented by the contractor who conducted the calculations.

Experience

One of the main challenges for Nagra in quality management of safety analyses and the safety case has been the development and implementation of a rigorous data clearance process. A specific challenge here is the question of when to "freeze" data used in the work needed to compile a safety case. If, on the one hand, this is done early in the project, then it is likely that developments in a specific area (e.g. development of canister design) will progress noticeably until the time the documentation is handed in to the authorities and that the actual data used in the project does not correspond to the data published in the most recent documentation on, in this example, a specific RD&D project with a longer time perspective. Although such "discrepancies" may have negligible implications for safety, they may be misused and quoted out of context by specific stakeholders. If, on the other hand, data freezing is done late in the project, then there may be too little time for all the sub-projects that use this data to do their work in the thorough manner required to assure the needed quality.

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Morsleben repository – Interdependence of technical feasibility and functionality of geotechnical barriers and safety case development

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Introduction

Based on a selection procedure whereby ten existing mines had been taken into consideration, the Morsleben repository for radioactive waste (ERAM) was built in a former mine for potash and rock salt production. The specific concerns and objectives of a repository for radioactive waste could not be taken into account when the mine was built at the beginning of the last century.

Irrespective of this, altogether about 37 000 m^3 of low-level and intermediate-level radioactive waste was stored in several areas of the mine between 1971 and 1991 and from 1994 to 1998. In the scope of the ongoing licensing procedure, the safety of the "historically grown" repository needs to be demonstrated for the phase after it has been sealed.

Contents of the proof of long-term safety ("safety case")

In the safety case for the ERAM (Wollrath, 2008, 2009), it has to be evidenced that no harmful effects for the biosphere need to be feared for the long term. For this purpose and on the basis of today's knowledge of the site and its possible development, a safety concept has been developed and evaluated with the help of which the requirements are to be fulfilled. Natural analogues, model calculations, laboratory and *in situ* tests and different forecasting techniques will be applied for the necessary, comprehensive safety demonstrations, which will also have to take into account the unavoidable uncertainties and knowledge gaps in terms of the site conditions and the long periods under consideration.

Systematic approach

In a systematic approach, the first step is to describe the site with the stored waste in detail. Apart from the description of the actual state and the possible developments of the site to be derived from this description, a repository concept also contains the technical components whose characteristics are to ensure the long-term retention capability of the geological and geotechnical barriers. For the ERAM site these are in particular the supporting effect of the backfill and the hydraulic resistance of sealing structures. The supporting backfill serves to maintain the integrity of the geological barrier in order to prevent for the long term an inflow of solution from the overburden. In case a significant inflow of solution should nevertheless occur, sealing structures are constructed with the objective to delay the contact between the solutions flowing into the mine cavities and the waste for as long as possible.

The major part of the periods of time considered in the long-term safety assessment cannot be measured and are thus not objectively verifiable. Therefore, the understanding of the system needs to be developed to the extent that the system behaviour for the long period of time under consideration can be forecast with sufficient accuracy on the basis of today's knowledge and measurements.

For the decommissioning of the ERAM – as with other repository projects – detailed planning and the developments in structural engineering have been made parallel to the development of the safety case. Therefore, input parameters are needed for calculations which are available already at an early planning stage, in order to implement the first estimations of the system behaviour and to build up an understanding of the system. For this reason one is forced to start planning based on plausible assumptions, the reasons behind which must be documented in the safety case.

ERAM is a complex system with non-linear system behaviour. Because of that, it cannot be determined in advance for each parameter used in the safety assessment how its modification will affect the overall system behaviour. Therefore it follows that it is often not possible to determine exact requirements on individual components of the decommissioning concept and thus on individual parameters in order to derive the success or failure of the safety case. It is therefore reasonable to carry out probabilistic model calculations – in addition to deterministic calculations – in which the parameters vary within defined ranges, in order to then be able to evaluate the resulting effects. Especially for the ERAM, the studies on system behaviour have shown that uncertainties and resulting weaknesses in individual components are compensated by the resulting reaction of the overall system. Thus, based on the safety concept, a decommissioning strategy has been developed that represents a robust system in terms of its retention capability for the disposed radionuclides.

Proof relating to geotechnical barriers

In the scope of the safety case, evidence of the sealing structures' functionality needs to be furnished, in addition to the proof of the salt barrier's integrity. The following statements are restricted to the first proof.

In the decommissioning concept, the sealing structures take over the function of delaying the contact between solution and waste for as long as possible in case of a relevant inflow of solution. As it cannot be reliably forecast whether an inflow of solution into the mine cavities will occur and, if so, when it will occur and at what rate, and what the composition of the solution at the sealing structures will be, solely the initial state of a sealing structure built according to plan can be measured (Mauke, 2013) and used for the furnishing of proof. With the help of suitable forecast models the characteristics of the sealing structures have then to be forecast location-specifically for the period under consideration.

However, the following considerations should be weighed in the determination and interpretation of the processes and parameters used in these forecast models:

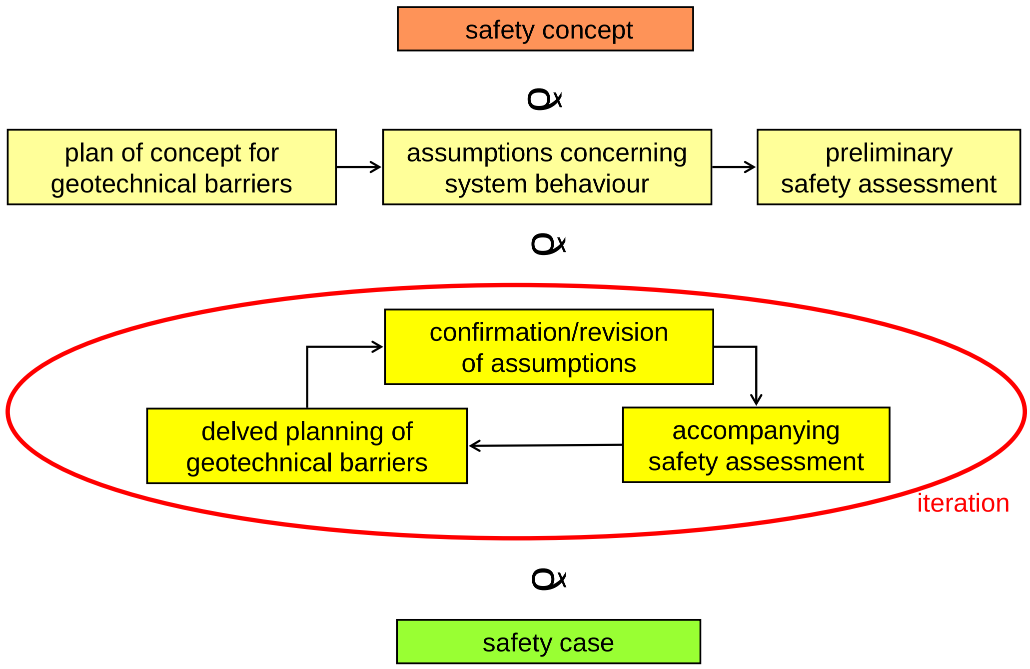
- Forecast reliability is significantly influenced by input parameter quality.
- The structures' characteristics can be influenced by the measuring equipment or the measurements themselves.
- The target value for the parameters to be determined is frequently in the range of the limits of detection of the measuring systems used.
- Within a realistic period of measurement, no stationary behaviours are yet to be expected. Therefore the extrapolation quality is subject to uncertainties.
- Not all possible system states (e.g. increase in pressure with different, particularly slow rates, corrosion effects in the case of different compositions of solutions, influence of saturation,...) can be simulated *in situ*.

Taking into account all these aspects it becomes clear that the measurements gained in the scope of *in situ* tests contribute essentially to the understanding of the system and thus also to the improvement of forecast models. A direct evaluation of the functionality or, respectively, confirmation of originally made model assumptions is only reliable after a comprehensive interpretation has been carried out. As a key component, the discussion of the interpretation results has to be taken into account in the documentation of the safety case.

Iterative approach

Ultimately, deviations from the assumptions originally taken into account in the safety assessment may result from the outcomes of the exact planning of the decommissioning measures and the *in situ* tests performed. Then a decision needs to be made as to whether account must be taken of these deviations by modification of individual elements of the decommissioning concept or by adaptation of the safety assessment, or both.

Figure 1: Sketch of the iterative approach



Example

A result of the detailed planning for the safety demonstrations relating to the ERAM sealing structures is given as an example. To provide redundancy for the case which cannot be ruled out completely that solutions from the overburden migrate into the mine, sealing structures are to be built at 22 locations. In the safety demonstration by way of calculation for these sealing structures, the geological conditions are a key input variable. For concept planning, evaluations from rock-mechanic modelling were used to derive input parameters for the sealing structures, to prove the suitability for use and to determine the structures' characteristics for a representative sealing structure. These results are contained in the safety case. Once the individual locations had been made accessible, the rock characteristics were examined by additional measurements in the scope of the detailed planning. It showed that at some individual locations the measured rock characteristics deviated from the original assumptions.

The BfS is currently working on adapting the construction engineering works for the sealing structures and the related safety demonstrations to the currently known information of the individual locations. Following this, it must be evaluated whether the entire safety case has to be adapted to the new findings. That is the case when key assumptions relating to the sealing structures' system behaviour are no longer in compliance with the fundamentals of the safety case.

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Sensitivity analysis: Theory and practical application in safety cases

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Sensitivity analyses – What are they and what are they for?

The target of an uncertainty analysis is to characterise as accurately as possible the uncertainty of model outputs (e.g. indicators for consequences associated with a repository system) resulting from uncertainties concerning the model input. Another key task in a performance assessment is to investigate the relationship between input and output behaviour by identifying the most relevant input parameters, which is known as sensitivity analysis (SA). Note that the problem setting is twofold: A description of input uncertainty is needed, as well as a simulation model representing physical properties. In accordance with Saltelli, *et al.* (2000) we understand by SA addressing both aspects and considering the input uncertainty given by probability distributions. The question is then how to define "relevant" or "important" or – in contrast – "irrelevant" and "unimportant" input parameters (Bolado Lavín, *et al.*, 2008). An input parameter can be considered important with respect to a given output variable if a strong correlation exists between both (linear relation), but it could also be considered so if the output is functionally dependent on the input or if that input contributes a large fraction of the output variance (considered a measure of importance). However, there is a real danger of misclassification by declaring an input parameter as being non-influential solely based on a linear regression or a rank regression analysis.

Another issue that arises in the sensitivity analysis area is the study of interactions. We say that a group of input parameters interact when their joint effect is different from the sum of their individual effects. Interactions deserve to be studied in order to identify the structure of the system model under study. Not all SA techniques are able to study interactions and in some cases, though they are able, the study could be impractical due to different reasons (extreme computational cost, too-large diversity of possibilities, etc.).

The existence of different interpretations of importance has triggered the development of a variety of SA methods designed to study the model from different points of view, each one developed according to a given interpretation. Nowadays a large corpus is available to the SA practitioner, who may choose appropriate methods to perform a specific type of SA attending to his/her interests and needs.

SA methods may be divided into three broad types: Local methods, screening methods and global methods. Local methods focus on the study of the system model behaviour under very specific system conditions (in the vicinity of a reference point), while screening methods focus on the functional relation between inputs and outputs disregarding input parameter distributions, and global methods focus on how the whole input space (taking input distributions into account) maps into the output space. Though all of them are important and provide relevant information about the system model, we focus in the following text on global methods.

To address sensitivity of the output on the input parameters several global sensitivity methods have been developed. Methods based on linear correlation or regression have been in use for decades, but they are not adequate for finding and correctly quantifying all kinds of input-output dependency. Therefore, more sophisticated sensitivity analysis methods have been developed. Among these, variance-based methods have been the most widely studied both from the theoretical and numerical viewpoints.

For practitioners, computing the R^2 coefficient of determination of a linear regression model is a widespread technique for assessing the reliability of linear sensitivity measures. But if non-linear and non-monotonic effects dominate the system's output behaviour then an R^2 measuring the goodness-of-fit for a non-linear regression model (using only a subset of the input parameters) is the next logical step. This sensitivity technique is known under many names: Pearson correlation ratio, Sobol' sensitivity effect, variance of conditional expectation, etc.

However, this approach requires that uncertainty be characterised sufficiently well by the variance: Consider a model producing a risk estimate ranging over several orders of magnitude. In this case, the uncertainty can often be described more naturally by the variation of the log-transformed risk, and not by the variation of the risk itself, as the exponential scale renders the variance useless.

To avoid such a dependency on a specific output transformation, distribution-based indicators are being developed. These take the average of a suitable distance between the unconditional output distribution and the output distribution conditional to a specific input parameter value. The metric can be chosen such that the indicator value stays constant, regardless of applying a monotonic transformation to the output.

Methods providing graphical feedback are advantageous to methods producing just a numeric output. Hence graphical sensitivity analysis methods provide hands-on results. Both distribution- and variance-based methods offer visual impressions.

Performing and applying sensitivity analysis in recent safety cases

Despite the wide range of available SA methods which are suitable for identifying and studying different types of input-output relationships, recent safety cases usually apply only a few. For obvious reasons, extensive straightforward deterministic (i.e. local) analyses of the impact of model assumptions, but also of input parameter choices (screening), typically make up a large part (Andra, 2005; SKB, 2006). In addition, linear or rank-based methods are often used for integrated safety assessment models (OECD/NEA, 2012a), safety indicators such as annual effective dose or annual individual risk being the output entities of concern. In the EU project PAMINA (2009), these and many other methods were studied, tested and further developed by RWM and research organisations, several of which were or are producing safety assessments and safety cases. However, these other methods are hardly ever used in "real" safety assessments, despite being able to detect sensitivities (e.g. non-monotonic effects or interactions) which will remain hidden when applying the more widely used linear or rank-based methods. The recent Swedish safety assessment SR-Site (SKB, 2011) goes a bit further by applying, in addition to rank regression, "a tailored regression model, based on the mathematical model used in the risk calculation." Dependent on the regression model, such a method has the potential of identifying various types of sensitivities. This regression model, however, relies on *a priori* knowledge concerning the risk calculation model and the question arises whether it will be able to reveal *a priori* unknown effects.

At the 2012 topical session of the Integration Group for the Safety Case (IGSC), the role of SA in the safety case was discussed. It was observed that sensitivity analyses can contribute to confidence building by confirming what was assumed about sensitivities or the lack thereof, hence using SA as a verification tool. Indeed, experience shows that SA can help identify hitherto undiscovered bugs. Moreover, it was claimed that instead of identifying R&D needs, SA results may support the safety case by confirming that uncertainties are *not* sensitive with regard to safety.

Further, the point was made that the established linear or rank-based methods often serve their purpose well. In the opinion of the authors of this paper, this is true as long as it can be shown that the identified sensitivities indeed explain the behaviour of the output (e.g. by means of the R^2 coefficient of determination). Also, if a model does not account for a relationship existing in the "real" system to be modelled (e.g. a non-monotonic one or a parameter interaction), obviously sophisticated methods capable of identifying such a relationship are of no use.

When dealing with disposal in rock salt, the above statement about the sufficiency of linear and rank-based methods appears to be inappropriate: The system is characterised by non-monotonic and discontinuous release behaviour, and these features are reproduced in performance assessment models.

Thus, the question remains about an appropriate and systematic SA strategy: Is there a "universal" sequence of applying various SA methods which makes it unlikely that sensitivities remain hidden and which, at the same time, does not waste staff and computer resources? IGSC's MeSA project (OECD/NEA, 2012b) suggested "to develop guidance on a general scheme for performing sensitivity analyses in safety assessments for geological disposal systems and interpreting results."

Such a strategy or scheme has to be capable of identifying various types of sensitivities, including:

- non-monotonic relationships;
- effects on probability density;
- interactions;
- threshold behaviour and discontinuities.

Since input parameters are not always independent, the question about how to address dependent input parameters also arises. Finally, the variation over time which is typical of many model outputs (e.g. safety indicators) must be appropriately addressed.

Going beyond linear and rank-based methods: The MOSEL and NUMSA projects

In order to build confidence in the safety assessment, it is advantageous to have a state-of-the-art sensitivity analysis concept available which provides undisputed and well interpretable sensitivity statements. The MOSEL and NUMSA projects, which are carried out by GRS and TUC and sponsored by the German Federal Ministry of Economics and Technology (BMWi), aim at developing such a concept applicable to different types of host rocks, associated repository concepts and corresponding safety assessment models.

The MOSEL project is compiling a comprehensive overview of global SA and sampling methods. The methods are classified and their applicability for a safety case is assessed. Therefore several models and scenarios (considering disposal in both salt and clay) for assessment and comparison of SA methods are defined as test cases. Several methods for sampling and SA are tested in view of the particularities of the test cases. Methods identified as suitable for application in the long-term safety analysis will be developed further, if necessary.

The NUMSA project is compiling the theoretical background of numerically efficient SA methods. Numerical efficiency is verified and assessed with test cases. Promising methods will be developed further, if necessary. Furthermore, methodical SA guidelines for time-dependent simulation results will be developed. The potential of meta-models will be investigated.

The safety assessment model currently considered within the NUMSA and MOSEL projects represents an LLW/ILW disposal site in rock salt, established in an abandoned salt production mine. The model resembles that for an existing facility of this type, showing some of its typical properties in a similar way, but is less complex. It is composed of two disposal chambers, one shaft, an intermixture chamber and a big compartment representing the residual mine.

Hands-on experience gathered in the course of the projects, in particular by studying the LLW/ILW model mentioned above, suggests the following procedure for performing SA: In advance of performing SA, it has to be conceptualised according to the assessment context; the model components to be analysed and the input and output entities of interest have to be identified and the associated uncertainties have to be described. Taking regulatory requirements as well as other issues to be addressed by the model(s) under question (e.g. interest in safety functions) into account, subjects of interest have to be acknowledged and the SA has to be conceptualised accordingly.

The LLW/ILW model is built of a probabilistic near-field module and deterministic geosphere and biosphere modules. The overall output can be seen as a sharp “probabilistic” image of the near-field modified by two blurring lenses, so it may be worthwhile to analyse the near-field output individually. According to this photographic analogue, SA methods analysing sharp details rather than the whole image can provide more information on specific facts, i.e. different behaviour of single radionuclides. Further investigations of this fact will be part of future research.

The SA itself should start with a dissemination of the available dataset and the system behaviour, bearing the assessment context in mind. This process should start with simple graphical methods, i.e. histograms, time-dependent quantiles, scatterplots and superimposed evolutions in time. Generally plots have to be interpreted carefully, e.g. time-dependent quantile plots do not have a matching realisation. More generally, the brain tends to construct structures, even from random patterns, an effect well known from optical illusions.

Cobweb plots produce one curve for every realisation so that the visual interpretation becomes more difficult with an increasing number of simulation runs. Mean ranks plots (Cormenzana, 2012) provide a handy solution to this problem by accumulating and averaging the output. The exemplary mean ranks plot (Figure 1) of the LLW/ILW model provides the same qualitative information on parameter importance as the Pearson correlation or the Spearman rank correlation. Color-coding may help to visually interpret plots, and here it gives a hint whether the parameter output relation is proportional or anti-proportional, or if only a part of the parameter space is important. Care must be taken not to over-interpret the details, as the general pattern is much more important than single lines.

In order to obtain quantitative rather than qualitative information (as provided by mean ranks plots) alternative methods have to be used, especially for ranking parameters by sorting them according to descending importance instead of arbitrary judging. Pearson correlation coefficients (PCC) and Spearman rank correlation coefficients (SRCC) are prevalent methods for estimating global sensitivity indices, considering only linear or monotonic trends, respectively. In contrast, variance-based sensitivity methods capture non-linear functional dependencies. However, one loses the sign information of PCC or SRCC signalling co-monotonic or contra-monotonic behaviour. Figure 2 shows the squares of PCC and SRCC as well as variance-based indices calculated by two different algorithms, EASI and COSI.

Figure 1: Mean ranks plot of LLW/ILW cumulative dose (2 000 runs, quasi-Monte Carlo sampling)

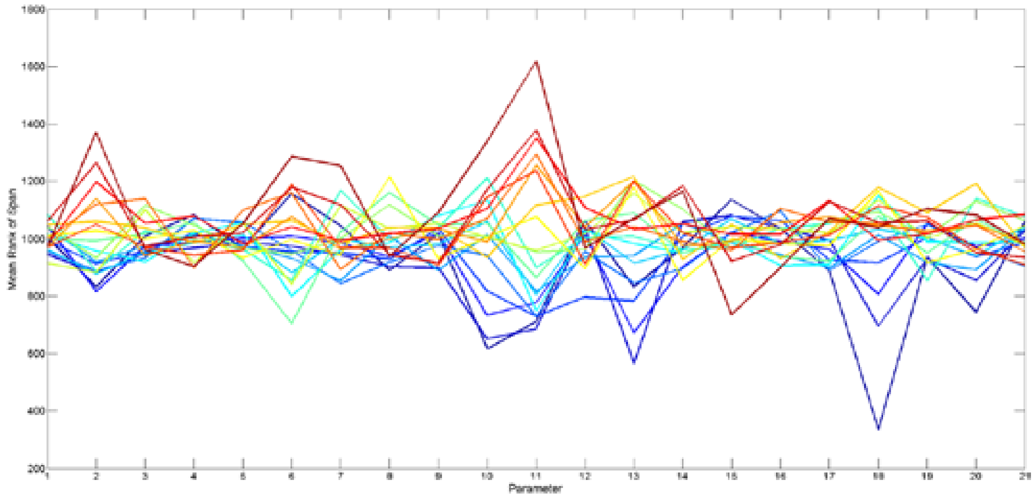
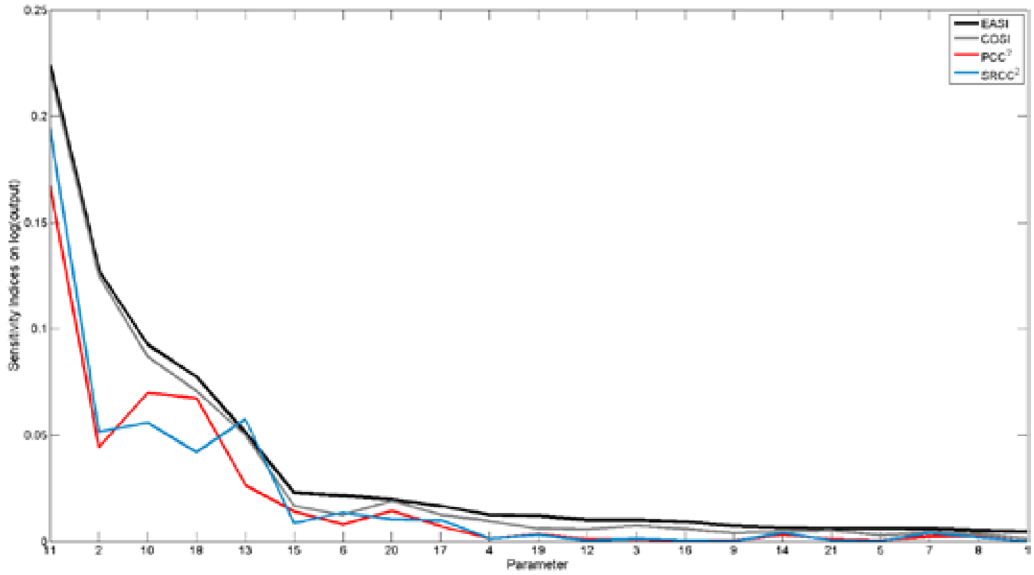


Figure 2: Comparison of different methods for estimating sensitivity indices (reordered parameters)



By squaring the PCC and SRCC comparability to the sign-less variance based indices is achieved. Parameters have been reordered with respect to descending EASI results. Here one can notice that the variance based indices capture information not provides by PCC and SRCC (parameter 2 and also 10, 11, 18).

As explained above, there is no single SA method capable of identifying "all" sensitivities of interest (irrespective of the nature of the model) efficiently. In contrast, the choice of the "best" SA method strongly depends on the nature of the model under consideration. Since this nature may be *a priori* unknown or not fully known, we suggest a stepwise approach, starting with the most simple and inexpensive methods.

Thus, the complexity level of SA should increase with increasing complexity of the model's behaviour as long as this behaviour is not sufficiently explained. For linear or monotonic models, graphical methods, Pearson correlation or Spearman rank correlation

(Saltelli, *et al.*, 2000) will yield robust and useful results. Complex models with non-linear non-monotonic behaviour require more elaborate methods, as simple methods cannot explain large parts of the model's behaviour. After each step (i.e. after each application of a specific SA method), the analyst has to ask whether the identified sensitivities explain the model behaviour sufficiently well by calculating a related goodness-of-fit indicator. If so, the SA can be considered complete. If not, the next step has to be carried out by applying a more sophisticated SA method.

Such a sequence of increasingly complex methods might include the following:

- As explained above, the analysis should always start with graphical/screening methods. Under favourable conditions, these will already yield all the information one is looking for. However, this cannot be proven since such methods do not offer any goodness-of-fit measures by themselves.
- Thus, one should proceed by applying linear (Pearson) methods. The R^2 coefficient of determination will indicate whether or not the model can be sufficiently explained by linear input-output relationships.
- Rank-based (Spearman) methods, being able to detect monotonic relationships, follow as the next logical step. Analogously, the rank-based coefficient of determination R^{*2} will indicate whether their application is sufficient.
- Up to now, our list only includes methods well established in safety assessment for radioactive waste disposal (perhaps with the exception of some graphical methods). Indeed, moving from Pearson and Spearman methods to variance-based approaches implies a change in underlying theory and numerical approach. Many, but not all, variance-based methods require specific and extensive sampling schemes, while Pearson and Spearman methods can be carried out based on any sample used already for probabilistic uncertainty analysis. There are exceptions such as EASI and COSI, which allow calculating variance-based measures using existing samples (Plischke, 2010, 2012). Naturally, one would start by calculating first-order sensitivity indices which describe how much of the output variance can be attributed to the variance of each single input parameter. Provided that the inputs are independent, a sum of these indices close to unity signals that the model behaviour is sufficiently well explained.
- If this is not the case, the reason might lie in the existence of parameter interactions. These can be identified by calculating variance-based higher-order sensitivity indices. However, not all methods are equally suitable to calculate higher-order indices. Again, in the case of independent inputs, if the sum of all calculated indices (first and higher order) is close to one, the SA can be considered completed.
- If the assessor is not yet satisfied, he/she might try performing transformations (e.g. rank or normal score transformations) and applying variance-based methods to the transformed data. Again, the sum of the calculated indices will provide information about the appropriateness of the method (given that the inputs are independent). However, one has to be aware that the transformations will hide information which was present in the original data sets (the same being true for the rank transformation used for Spearman methods, by the way). The use of transformations for variance-based methods is under investigation in the MOSEL/ NUMSA projects.
- If all the above does not lead to satisfying results, distribution-based methods should be applied. Here, judging the degree to which the result explains the model behaviour is based on a different reasoning: If the expected shift of the output distribution when fixing one or more input parameters then there exists no significant dependency (Plischke, *et al.*, 2013). Distribution-based methods are currently being further investigated.

Summary and outlook

The projects described here aim at deriving an adaptive and stepwise approach to sensitivity analysis (SA). Since the appropriateness of a single SA method strongly depends on the nature of the model under study, a top-down approach (from simple to sophisticated methods) is suggested. If simple methods explain the model behaviour sufficiently well then there is no need for applying more sophisticated ones and the SA procedure can be considered complete.

The procedure is developed and tested using a model for a LLW/ILW repository in salt. Additionally, a new model for the disposal of HLW in rock salt will be available soon for SA studies within the MOSEL/NUMSA projects. This model will address special characteristics of waste disposal in undisturbed rock salt, especially the case of total confinement, resulting in a zero release which is indeed the objective of radioactive waste disposal. A high proportion of zero-output realisations causes many SA methods to fail, so special treatment is needed and has to be developed. Furthermore, the HLW disposal model will be used as a first test case for applying the procedure described above, which was and is being derived using the LLW/ILW model.

How to treat dependencies in the input, model conservatism and time-dependent outputs will be addressed in the future project programme:

- If correlations or, more generally, dependencies between input parameters exist, the question arises about the deeper meaning of sensitivity results in such cases: A strict separation between inputs, internal states and outputs is no longer possible. Such correlations (or dependencies) might have different reasons. In some cases correlated input parameters might have a common physically (well-)known fundamental cause but there are reasons why this fundamental cause cannot or should not be integrated into the model, i.e. the cause might generate a very complex model which cannot be calculated in appropriate time. In other cases the correlation may be supported by empirical data, for which no (satisfactory) explanatory theory exists.
- Conservatism in modelling and/or parameter assumptions has an impact on SA. Awareness is needed that SA is addressing the simulation model and not the physical system under consideration. Conservatism in itself is a mixed blessing, it can help handling lacks of knowledge but over-conservatism leads to heavy loss of conclusiveness.
- In the case of time-dependency, especially in simulation over time periods of up to a million years, it is expected that a change in parameter importance will occur over such a long time. The question is how to identify, treat and rate time-dependent sensitivity indices.

The authors of this paper wonder whether the possibilities and capabilities of SA are presently under-used in safety cases or whether appropriately performed SA might contribute to improved model (and system) understanding and to the identification of possibilities to strengthen model robustness by identifying R&D options. This issue, however, goes far beyond the scope of the projects described here.

It is also remarkable that in most, if not all, cases global SA methods have been applied to integrated models and safety indicators such as annual effective dose or annual risk. In the projects described here, SA is additionally being applied to flux-related performance indicators, by such means focusing the interest on one sub-system rather than the repository system as a whole. The authors also wonder about the feasibility and potential benefits of applying SA to process models. Such often complex and non-linear THMC models generate outputs which could be interpreted as safety function indicators (OECD/NEA, 2012b).

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Session 6.2

Specific Issues and Challenges in Safety Case Development, Part II

R&D in safety case development – The contribution from the Implementing Geological Disposal of Radioactive Waste Technology Platform (IGD-TP)

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Introduction

The Implementing Geological Disposal of Radioactive Waste Technology Platform (IGD-TP) was launched in November 2009 to identify and implement the remaining research development and demonstration (RD&D) activities in Europe in the area of deep geological disposal of high-level and other long-lived radioactive wastes. The work focuses on the safety and demonstration of the technologies needed for implementing deep geological repositories, and in underpinning the development of a common European view on the main issues related to the safe management and disposal of radioactive wastes.

The European waste management organisations share the opinion that it is time to proceed to license the construction and operation of deep geological repositories for spent fuel, high-level waste, and other long-lived radioactive waste. The IGD-TP's "Vision 2025" is "that by 2025, the first geological disposal facilities for spent fuel, high-level waste, and other long-lived radioactive waste will be operating safely in Europe." (IGD-TP, 2009) IGD-TP also fosters co-operation and transfer of technologies, which are both beneficial to an increase in efficiency and cost-effectiveness of RD&D. In order to reach this vision, a strategic research agenda (SRA) was compiled and will be updated when needed. The SRA provides an overview of the remaining RD&D priorities which are, on the one hand, key in moving towards implementation of geological disposal (and thus establishing safety cases) and, on the other hand, suitable for co-operation at the European level. The SRA is an important document for communicating the remaining research needs, but also an instrument for creating synergies, co-operation and co-ordination with activities taking place in other technology platforms and within other international co-operation forums.

Development stages of the European WMO's and the related RD&D

The "Vision 2025" of the IGD-TP can only be achieved through the progress of individual waste management programmes towards the implementation of geological disposal. At the moment the "Vision 2025" is within reach in a few European Union member states.

In some programmes a longer period is still required for preparation, whilst others are still at an early stage of development. Therefore, it is natural that the recognised RD&D needs of those programmes closest to licensing receive particular attention in the context of the SRA. Approaching the granting of a license illustrates that enough confidence is available to allow taking decisions in the stepwise development and implementation of a geological repository, but does not mean that no more research is needed. However, this may lead to a shift in research domains and priorities in order, notably, to take into account remarks from regulators and to prepare for industrialisation of repository construction and disposal operations. Most probably, even during operation, RD&D activities will remain necessary, for example, to continue to optimise the system, assess monitoring results, test new devices, etc.

Some waste management programmes are at early stages of development or have longer time schedules and/or are subject to changes of their political situation (e.g. new European member states). Developing and implementing these programmes may require appropriate expertise and infrastructure in addition to their national regimes (e.g. national decision-making frameworks). Basic research related to geological disposal and education and training may also be of higher priority compared to programmes closer to licensing.

For these reasons, the IGD-TP has emphasised those issues that are material for reaching the vision of operational radioactive waste repositories by 2025. Nonetheless, even for programmes with later implementation dates, the nature of RD&D activities for any given stage, as well as the sequence of stages, are expected to be similar. Thus the results achieved by the programmes close to licensing will be of benefit to all other programmes. For the same reasons, some of the RD&D issues that may be of key importance for the achievement of objectives in some individual programmes are of lower common interest to the participating WMOs, if they are specific to individual programmes. For example, this is often the case with site characterisation and the interpretation of its results, and therefore, even if highly important for geologic disposal, this area is discussed only briefly in the SRA.

On a similar sort of reasoning, the SRA also does not discuss the detailed design and the licensing processes that are individual waste management programme specific issues. However, in some cases host rock specific RD&D issues are of general importance with added value to other geological disposal options as well.

National RD&D needs

The EC Directive 2011/70/Euratom defines that all member states shall present national programmes for the management of spent fuel and radioactive waste. Such programmes, including RD&D needs, already exist in many European member states, though in different forms owing to national regulations and internal needs, and are in some cases revised at regular intervals.

The RD&D plans at the national level might differ significantly as they strongly depend on the national context (national laws, stage of the programme, type of host rock considered, stakeholder interactions, etc.). Each waste management organisation (WMO) focuses on carrying out RD&D that helps to deliver the input, answers and state of the art needed for the next programme stage and beyond, based on the available information and knowledge within the geological disposal community.

Based on the context (including available host rock geology), this is translated into technical and safety requirements for specific components. Here, other boundary conditions might also be relevant, for example specific questions from the regulator. The resulting evaluation then leads to an RD&D plan that is adequate to perform the system development and assessment needed for the next programme stage. Within this approach for developing RD&D plans and system evaluation, the needed interaction between safety, design and process understanding is considered.

In accordance with international guidance the aim of the long-term management of high-level and/or long-lived radioactive waste is to protect "man and the environment, now and in the future". At the international level, there is a consensus that the maximum level of passive safety can be obtained through geological disposal. The disposal system consists of engineered and natural barriers between the wastes and the surface environment in order to prevent radionuclides and other toxic species reaching the surface in such concentrations that they could present an unacceptable risk.

The safety concept describes the conceptual understanding of why the disposal system is safe. The disposal system performs the broad safety functions via a range of features and associated processes that vary in their effectiveness and in the level of scientific and technical understanding that is available. This safety concept is the main starting point to define the technical and scientific requirements for the disposal system and its specific components. However, other boundary conditions might intervene, like for example specific questions from the regulator or other stakeholders (e.g. specific questions on reversibility and retrievability). The safety concept is built on a limited number of effective and well-understood features that ensure that the disposal system is safe and that safety can be demonstrated, even allowing for the various uncertainties and harmful events and processes that might affect the system's evolution.

On this basis RD&D plans are developed that address the need for further scientific and technical knowledge that is required to carry out performance and safety assessments and the integrated safety case before proceeding to the next programme stage. In practice, this involves iteration between design options, demonstrable performance and continued research into process understanding of the [chosen] disposal system. During such iteration, estimates of performance are made and an understanding is developed of which elements of the disposal system actually provide safety under various conditions, thus refining the disposal concept.

Development of a joint strategic research agenda

The identification of key topics for the SRA started with inputs from participating WMOs using individual host rock specific safety cases and associated RD&D programmes.

Starting with the identification and selection of potential issues to be addressed in the SRA, the participating WMOs provided a first view on the kind of products needed, the improvements of present approaches and the importance to the respective waste management programmes in general. The RD&D issues considered by the individual member organisations are, to some extent, dependent on the host rock options pursued and the specific disposal concepts developed.

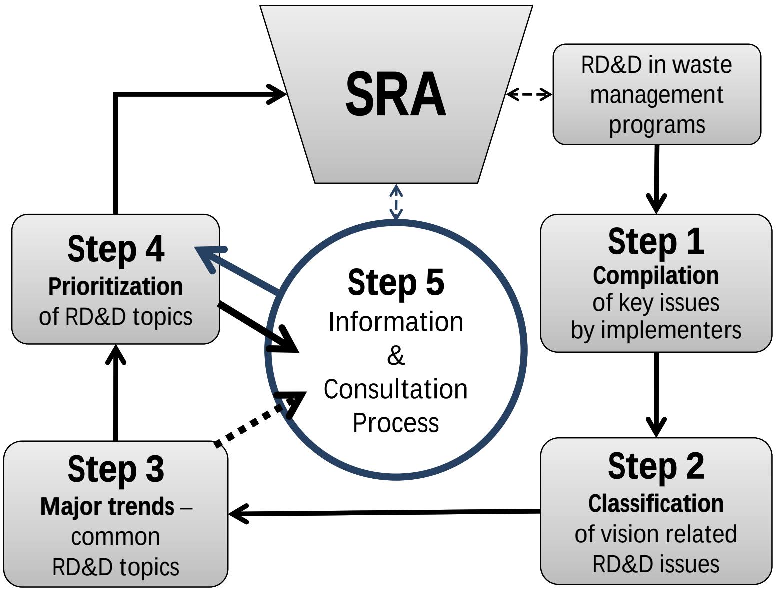
Due to the mature state of both the understanding of the geological disposal systems as well as the development of emplacement technologies and safety-related components the issues generally fall in one of the three main categories:

- demonstration of long-term safety;
- development and demonstration of disposal techniques and components;
- site characterisation and confirmation of site suitability.

Regarding the role of underlying scientific research in the SRA, more than 30 years of such studies have led to a strong scientific basis for geological disposal. Nonetheless, where required such work will continue and, where sufficiency of information exists and no further RD&D is actively pursued, the scientific basis of the safety case will need to be continuously updated. As this underlying scientific research is a broad requirement intrinsic to several key topics of the SRA, this aspect has not been directly highlighted.

The members of the IGD-TP have adopted a systematic and well-structured strategy for the development of the SRA, which follows a staged process basically made up of five consecutive steps (IGD-TP, 2011), as illustrated in Figure 1.

Figure 1: SRA development



This stepwise procedure led to the identification of seven main thematic areas that were defined as key topics:

- safety case;
- waste forms and their behaviour;
- technical feasibility and long-term performance of repository components;
- development strategy of the repository;
- safety of construction and operations;
- monitoring;
- governance and stakeholder involvement.

The goal of the deployment of the activities flowing from the SRA is to assist the IGD-TP members in achieving the vision and the desired results by joint RD&D activities during the next years (IGD-TP, 2012). The platform intends to constitute means to further build confidence in the solutions, to reduce overlapping work and avoid duplication of existing forums, to produce savings in total costs of research and implementation, and to make better use of existing competence and research infrastructures. This is done e.g. by pooling critical resources and preparing co-ordination of future projects, and also by pooling resources for other types of joint activities.

In line with the SRA key topics and the deployment plan (DP), IGD-TP has currently set up 12 out of 16 working groups related to joint activities. Five of them have convened at least once in the first half of 2013. In addition, 7 EU projects, launched after creation of the IGD-TP, are followed up by members and their progress and outcomes are reported three times a year during Executive Group meetings. Three groups are actively preparing proposals which will be ready for the future first H2020 call. We expect to launch in the forthcoming years a minimum of two EU technical projects per year and at least one internal project.

The platform is also engaged in support of the Competence Maintenance, Education and Training group whose aim is to establish a coherent framework of training schemes aiming at preparing the shortage in engineers and researchers in the field of geological disposal that may occur in the future.

Finally, the platform is engaged in a process of dissemination of the IGD-TP activities and achievements. This has led to the development of a new website (www.igdtp.eu), an increasing participation in conferences organised to address, in particular, the needs and projects of the new member states and co-operation with the EU.

Transferability – How transferable?

As mentioned, given the vision of the IGD-TP, the recognised RD&D needs of those countries closest to licensing receive particular attention in the context of the SRA. However, all members are convinced that the IGD-TP contributes added value for all countries. First, it allows creating a common knowledge basis, which is a peer-reviewed, acknowledged foundation of geological disposal. Second, it should avoid duplication of errors made in the past. It is important to learn from past mistakes and failures and the platform allows sharing this information. Finally, through the platform, information can be transferred, and this can be very diverse in nature, going from PA/SA methodologies to experimental designs for e.g. *in situ* experiments.

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Harmonisation of regulations on back-end activities – WENRA

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Abstract

The Western European Nuclear Regulators Association (WENRA) was established in 1999. The main objectives at that time were to develop a common approach to nuclear safety in Europe and to provide an independent capability to examine nuclear safety in applicant countries.

Two working groups were launched to harmonise safety approaches between countries in Europe, the Reactor Harmonisation Working Group (RHWG) and the Working Group on Waste and Decommissioning (WGWD). In response to the events in Japanese reactors following the tsunami in 2011 WENRA established the contents of the "NPP stress test". Recent WENRA activities are concerned with inspection practices and research reactors.

The WGWD has to date developed Safety Reference Levels (SRL) reports for decommissioning and storage according to its original mandate (WENRA, 2011, 2012a). WENRA members have experienced a benchmarking process and established national action plans for the modification of their national legal systems and practices according to benchmarking results.

WGWD is currently working on developing a SRL report for disposal facilities for radioactive waste. A first draft version with SRLs for disposal was published in November 2012 on WENRA's web page, for comments from stakeholders. This paper presents the current status of development and elaborates on the role of WENRA WGWD work in harmonising approaches in Europe regarding development of the safety case for disposal of spent fuel and radioactive waste.

Introduction

There were two main reasons for the establishment of the Western European Nuclear Regulators Association (WENRA) in 1999. Firstly, nuclear safety was included in the European Union set of enlargement criteria, and secondly, national safety approaches have been developed from IAEA Safety Standards and the Convention on Nuclear Safety, though independently.

The main objectives of WENRA at that time were to develop a common approach to nuclear safety and to provide an independent capability to examine nuclear safety in applicant countries. In March 2003 the objectives of the co-operation within WENRA was extended, in addition to the objectives set out in 1999, to become a network of chief nuclear safety regulators in Europe exchanging experience and discussing significant safety issues.

Today, WENRA consists of 17 members[1] and 9 observers[2]. WENRA countries are committed to developing their national practices so that agreed SRL will truly be achieved in the regulatory system of each member country.

Two working groups were launched to harmonise safety approaches between countries in Europe, the Reactor Harmonisation Working Group (RHWG) and the Working Group on Waste and Decommissioning (WGWD).

The mandate of the working groups was to analyse the current situation and the different safety approaches; to compare individual national regulatory approaches with the IAEA Safety Standards; to identify any differences; and to propose a way forward to possibly eliminate the differences. The proposals were expected to be based on the best practices among the most advanced requirements for existing power reactors and nuclear waste facilities.

The harmonisation process in the field of reactor safety was the first to get underway; selected technical areas with identified needs for harmonisation were addressed.

WENRA WGWD activities

The Working Group on Waste and Decommissioning (WGWD) was established about two years after the RHWG and had to face a much greater variety of installations and tasks. This working group chose a holistic view on the safety of its facilities and practices rather than looking at a selected set of topics. Due to this completely different approach a first set of storage references, which was supplementing a copied set of reactor safety reference levels, turned out to be impracticable. Employing a rather time consuming second approach, WGWD finally managed to define comprehensive sets of SRL for decommissioning (of any type of nuclear facility) and storage (spent fuel and radioactive waste) (WENRA, 2011, 2012a). A first set of draft SRL for disposal was published in November 2012 on WENRA's web page for comments from stakeholders. The draft disposal SRL report is currently being reviewed and updated according to comments received.

Development of safety reference levels

The safety reference levels (SRL) are based on national safety requirements, corresponding IAEA Safety Standards, lessons learned from previous work to develop SRL, feedback from stakeholders and – last but not least – the personal experience of WGWD members.

Once the SRL are formulated, the countries have to go through a self-assessment process to evaluate the compatibility of their own regulatory systems with the SRL. In the case of storage SRL the operational practice in the relevant facilities has been benchmarked as well.

Benchmarking

The national benchmarking process is initiated after the realisation of self-assessment. During the benchmarking exercise, SRL fulfilment quotations from national regulatory

1. Belgium, Bulgaria, Czech Republic, Finland, France, Germany, Hungary, Italy, Lithuania, Romania, Slovak Republic, Slovenia, Spain, Sweden, Switzerland, the Netherlands, United Kingdom.
2. Armenia, Austria, Denmark, Ireland, Luxemburg, Norway, Poland, Russian Federation, Ukraine.

systems of each country are discussed and finally approved in subgroups of four to five WENRA countries. SRL are regarded as deficient (C-rating) unless they are clearly covered by existing national regulatory requirements (A-rating). On rare occasions justified deviations may be accepted (B-rating). Some typical examples for B-ratings are:

- In a small country one specific facility will be sufficient for the total nuclear programme, that facility already exists and the respective SRL is covered as a license condition.
- Due to the specific nature of the national management programme for spent fuel and radioactive waste, the respective SRL is not applicable.

National action plans

National action plans (NAP) are established for each country on the basis of its C-ratings in the benchmarked self-assessment. WENRA members strive to complete their national action plans by adjusting their regulatory systems usually within a period of two to three years, which is a quite challenging task. Countries who notify the completion of their NAP will undergo a re-benchmarking using the already familiar methodology, eventually with the result that they have implemented the full set of SRL in their regulatory system.

By end of February 2013, 12 out of 17 WENRA member countries had successfully carried out their NAP for storage SRL, with several other countries nearly finished. For the decommissioning SRL the same situation is expected one year later.

In addition to benchmarking against the regulatory systems, storage SRL were also benchmarked against factual implementation by implementers through a comparable assessment and benchmarking procedure. The most interesting results were:

- No single SRL already addressed in national regulations was missed in facility operation.
- In many cases SRL which were C-rated during the benchmarking of regulatory systems were already implemented in facility operations, due to e.g. peer review missions or as a result of the continuous improvement incentive of their own management systems.

Safety reference levels for disposal

For most types of nuclear facilities much input for developing SRLs can be extracted from existing national safety requirements, as they are based on extensive feedback of experience form regulating construction and operation of such facilities. The situation for disposal is rather different.

There is considerably less experience from construction and operation of disposal facilities, especially disposal facilities for spent fuel or high-level and/or long-lived waste. Thus, more emphasis has been placed on using IAEA Safety Standards as a basis for the development of disposal SRL, compared to development of SRL for non-disposal nuclear facilities.

Siting aspects for a reactor – or a storage facility – requires consideration of the operational time of the facility only, i.e. the safety assessment must consider all relevant influence from the chosen site on the operations of the reactor to prevent any accident from occurring. Also, there are no post-closure (post-decommissioning) aspects to be considered for siting and design of reactors or storage facilities.

Siting aspects have a very different implication on the post-closure safety for disposal facilities especially with regard to the extremely long time frames to be considered. The characteristics of the host environment are more or less crucial in order to provide for long-term containment and isolation of the spent fuel or radioactive waste disposed of.

A general challenge as regards development of SRL for disposal is that SRL are expected to be implemented in a member state’s regulatory system, and the requirements in the regulatory system are in general only applicable to licensed activities. Much of the needed research and development activities to develop a safety case take place before a license is applied for. Hence it might not be possible to enforce requirements during this pre-licensing phase, i.e. at the early stages of the development of a disposal facility. It is therefore anticipated in the disposal SRL report that whenever SRL are addressing a future licensee who is not yet subject to formal regulatory review, the licensee shall demonstrate fulfilment of these requirements at least when applying for the first licence.

Another challenge relates to the fact that disposal is to be seen as the endpoint of a long sequence of activities, from generation through conditioning, storage and transport to disposal, and quite frequently involves different operators and licensees. Thus, the conditioning and packaging of waste may have a large influence on post-closure safety. As individual licensees can only be responsible for activities within the envelope of their respective licence, it is important that the regulatory system ensure that interdependencies between different licensees are properly addressed in the overall regulatory system, i.e. to apply a cradle-to-grave approach.

Yet another challenge is that some disposal facilities are likely to be operated for many decades where construction of the facility and emplacement of waste as well as partial closure of the disposal facility may be carried out in parallel. Thus, the development of the safety case for a disposal facility and system presents specific challenges compared to the safety case for nuclear facilities of other types: it has to address post-closure safety in conjunction with operational safety. Further, the safety case matures as the disposal concept develops and the facility is constructed and operated.

Future activities

WENRA WGWD is currently working on reviewing and updating the draft disposal SRL report according to comments received by stakeholders. It is envisaged that the first version of the final report to be used for the benchmarking process against national regulatory systems will be finalised by WENRA in 2014.

Also, in order to cover all back-end activities WENRA foresees the development of SRL related to pre-treatment and conditioning of waste. The timeline for the development of such SRL is not yet defined.

As discussed above, disposal of spent fuel and waste is to be seen as the endpoint of a long sequence of activities, from generation through conditioning, storage and transport to disposal, involving different operators and licensees, where each licensee is responsible for activities within the envelope of their respective licence. In order to provide a proper context for WGWD work and to better understand the interfaces between the different WENRA WGWD SRL reports, WENRA also foresees the development of a general descriptive document. Such a document would also address the interdependencies between different licensees in a national context.

Concluding reflexions

WENRA members agreed to use the safety reference levels (SRL) as minimum requirements to be implemented in their regulatory systems. Nevertheless, they are free to further enhance their regulatory framework by introducing additional and/or more stringent requirements.

The concept for developing the SRL, i.e. benchmarking against national regulatory systems and the IAEA Safety Standards, contributes to harmonised safety approaches, including the development of safety cases. The participants of the WGWD are all involved in the elaboration of the national regulatory framework and also carry out inspections in their home countries. That results in a deep understanding during the elaboration of SRL and a fast and pragmatic implementation. This is the most unique and valuable property of the WENRA and especially the WGWD procedure. WENRA safety reference levels are genuinely respected documents, and the clear results and benefits are recognised by the stakeholders.

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The new ICRP recommendations on radiological protection in geological disposal of long-lived solid radioactive waste

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Radioactive waste management has been the subject of several recommendations of the International Commission on Radiological Protection (ICRP) since 1985. The aim of the new Publication 122 (2013) is to describe how the 2007 general recommendations of the Commission (Publication 103) can be applied in the context of geological disposal. For this purpose, it is important to emphasise that the new approach developed by ICRP is based on three types of exposure situations: planned, emergency and existing:

- Planned exposure situations correspond to situations where exposures result from the operation of deliberately introduced sources. Exposures can be planned and fully controlled.
- Emergency exposure situations correspond to situations where exposures result from the loss of control of a source within a planned exposure, or from an unexpected situation (e.g. malevolent event). These situations require urgent actions to prevent or mitigate exposures.
- Existing exposure situations correspond to situations where exposures result from sources that already exist when decisions are taken to control them. The characterisation of exposure is therefore a prerequisite for their control.

The application of the three basic radiological protection principles – justification, optimisation of protection and limitation of individual doses – are therefore considered in this new framework with justification and optimisation applying to the three types of exposure situations and limitation only to planned exposure situations.

The main points highlighted in Publication 122 for the application of the system of radiological protection to geological disposal of long-life solid radioactive waste are the following:

- For the protection of the future generations, the Commission's Recommendations are based on the fundamental principle that: "Individuals and populations in the future should be afforded at least the same level of protection as the current generation." (This principle was already in ICRP Publication 81.)
- The Commission considers that the potential exposures to humans and the environment associated with the expected geological disposal for long-life solid radioactive waste correspond to a planned exposure situation. They are taken into account at the time of the design of the disposal facility and protection strategies have to be developed to cope with these potential exposures.

- The application of the radiological protection system is significantly influenced by the level of oversight and "watchful care" of the disposal facility. Three major time frames have to be taken into account for radiation protection purposes:
  - the period of direct oversight during which the disposal facility is operated and is under active supervision;
  - the period of indirect oversight during which the disposal facility is partially or fully sealed and where indirect oversight takes the form of regulatory supervision, administrative and societal oversight;
  - the period of no oversight corresponding to a situation where the memory of the disposal facility has been lost and society is no longer in a position to keep a watchful eye on the facility; it is important to note that the Commission recommends to preserve oversight as long as possible because of its contribution to the overall effectiveness of protection.
- In this perspective, the various decisions to be made over time regarding the evolution of the monitoring and oversight should be discussed with stakeholders.
- Even when the memory of the disposal facility has been lost, leading to the abandonment of all forms of monitoring of the facility, the intrinsic functions of the repository still exist. The potential to retain and isolate the radioactive waste is an inherent feature of the repository and it continues into the distant future. This leads to address potential exposures as a planned exposure situation, except in the case of major disruptive events.
- For the application of the principle of justification, waste management and disposal operations have to be considered as an integral part of the practice generating the waste. In addition, it is recommended to review this justification over the lifetime of that practice whenever significant new information or an event might call into question the justification.
- The optimisation of protection is the central element for the implementation of the system of radiological protection in the case of a geological disposal facility. This approach, based on a stepwise process during the various stages of disposal (design, operation, monitoring) should enable the systematic and transparent assessment of protection options, including considerations on the best available techniques. The objective of the application of the optimisation principle is to strengthen the capacity of storage protection and reduce potential impacts (radiological and others).
- In accordance with the optimisation principle, the radiological criterion recommended by the ICRP for the design of a disposal facility is an annual dose constraint for the population of 0.3 mSv per year and a dose constraint below the annual dose limit of 20 mSv per year or 100 mSv over 5 years for occupationally exposed workers.
- A risk constraint for the population of 1×10^{-5} per year is also recommended when applying an aggregated approach combining the likelihood of the exposure scenario and the associated dose.
- It is recalled that the dose assessments and risk over the long term should be used for the comparison of options rather than a means of assessing health detriment.
- For natural events considered in the design of the disposal, the Commission recommends choosing dose or risk constraints in the band used for planned exposure situations.
- For severe natural disruptive events not taken into account in the design basis evolution as well as for the inadvertent human intrusion, the dose or risk

constraint does not apply. In that case, if the events were to occur during the period of direct or indirect oversight of the disposal facility, then the competent authority should take appropriate measures with reference to the emergency and/ or existing exposure situations.

- The implementation of the recommendations of the Commission requires a management system that integrates safety, health, environment, security, quality and economic considerations, stressing that safety remains the fundamental objective. The system has to support a transparent approach involving all relevant stakeholders.
- For the protection of the environment, in the absence of detailed assessment and corresponding criteria, evaluations in this area have to be addressed with a view to inform decision making.

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Sustainable network of independent technical expertise for radioactive waste disposal

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SITEX is a 24-month FP7 project led by IRSN and bringing together 15 organisations representing technical safety organisations (TSO) and safety authorities, as well as civil society outreach specialists involved in the "regulatory" review process of geological disposal of radioactive waste. SITEX aims at establishing the conditions required for developing a sustainable network of experts from various horizons (authorities, TSO, academic organisations, civil society,...) capable of developing and co-ordinating the technical expertise that is required from the stakeholders in charge of delivering opinion, independently from the waste management organisations (WMO), on the safety of geological disposals.

The SITEX programme of work is split into a set of six work packages that address technical and organisational issues allowing to propose a structure of the missions and operating mode of the future network. These issues relate on the one hand to the study of the potential for sharing and developing technical expertise practices amongst stakeholders, on the other hand on the ability to implement co-ordinated R&D programmes run by TSO in order to develop the scientific knowledge necessary to perform independent technical assessments. Two major perspectives are identified for the future of the SITEX network: its ability to foster co-operation between regulatory bodies, TSO, implementers and civil society with a view to enhancing common understanding of key safety issues and challenges and to identifying possible harmonisation of practices; the constitution of a scientific task force (mainly driven by TSO) for research definition and implementation at the European level allowing to improve the co-ordination of scientific programmes between TSO and developing its own skills and analytical tools, independently of the WMO.

A comprehensive list of safety issues relevant to the development and implementation of a geological repository has been developed. For each of these issues, the needs of national safety authorities and of TSO for dialogue and/or guidance development and harmonisation were identified.

A set of R&D activities was structured into main key safety issues related to the safety functions and components of the deep geological disposal. This will constitute the basis for the development and future implementation of a scientific research agenda (SRA) proper to expertise bodies and technical safety organisations. In parallel, an assessment of available technical means and scientific skills amongst SITEX partners is under consideration in order to identify areas for possible co-operation, exchange of staff and needs for improvement. This will contribute to assess how in practice this SRA could be implemented amongst organisation in close co-operation. Another objective of the elaboration of such SRA is to engage in scientific dialogue with IGD-TP, keeping in mind the context of the new EC H2020 programme and the identification of possible research projects of common interest for IGD-TP and SITEX.

A tentative framework for deriving a harmonised safety review method is proposed mainly based on development of existing material produced by the former European Pilot Study and the IAEA GEOSAF project.

As an integrated part of the technical and scientific review process, SITEX engaged with the identification of practical ways to interact with civil society in the expertise function on a durable process. For this purpose, a dedicated meeting was organised with actors of the civil society in Senec (Slovakia) in September 2013.

Finally, a framework document describing the characterisation of the national expertise function and elaborating a reflection on what could be the missions and the role of the future European SITEX network will be issued on the basis on the above elements with the objective to establish the future terms of reference of the network.

Session 7

Four Parallel Technical Sessions

Session 7.1

Performance and Safety Assessment

TURVA-2012: Performance assessment

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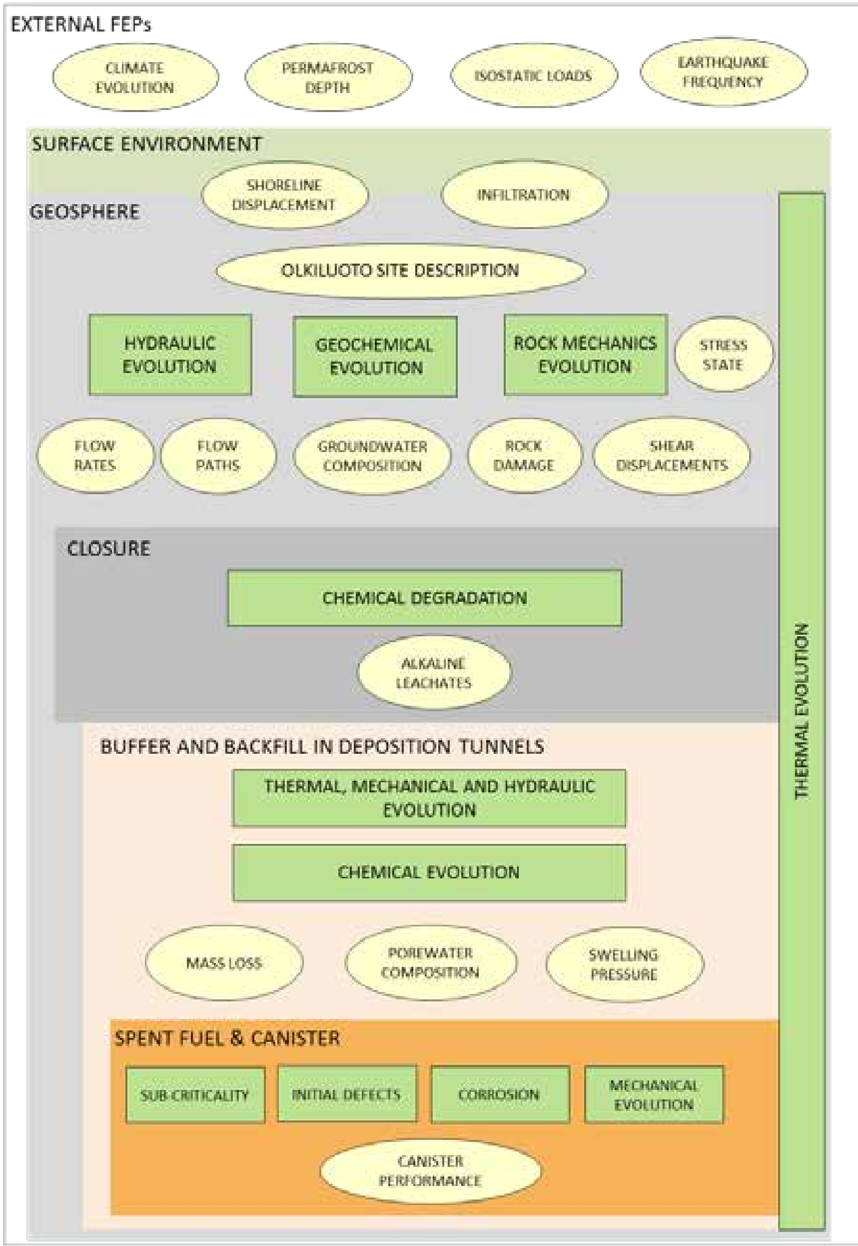
Introduction

TURVA-2012 is Posiva's safety case in support of the Preliminary Safety Analysis Report (PSAR) and application for a construction licence for a repository for disposal of spent nuclear fuel at the Olkiluoto site in south-western Finland. Posiva's safety concept is based on long-term isolation and containment, which is achieved through a robust engineered barrier system (EBS) design and favourable geological conditions at the repository site. The reference design considered in the TURVA-2012 safety case is the KBS-3V design, with the EBS consisting of a copper-iron canister, a buffer of swelling clay material, a backfill in the deposition tunnels of low-permeability material and closure of the central tunnels and other underground openings. The host rock acts as a natural barrier. Each barrier contributes to safety through one or more safety function. The conditions needed for the barriers to fulfil their respective safety functions are expressed in terms of performance targets for the EBS and the target properties for the host rock.

The performance assessment (Posiva, 2013), which is a key component of TURVA-2012, analyses the ability of the repository system to provide containment and isolation of the spent nuclear fuel during the long-term evolution of the system and the site. The conditions needed for the barriers to fulfil their respective safety functions are expressed in terms of performance targets for the engineered barriers and target properties for the host rock, for example properties related to the corrosion resistance and mechanical strength of the canister as well as groundwater flow and composition.

The analyses take into account the uncertainties in the initial state, the subsequent thermal, hydraulic, mechanical and chemical evolution of the repository system and uncertainties in the evolution. The conclusions of the performance assessment are based mostly on the output of key modelling activities shown in Figure 1. Whenever modelling is not possible the conclusions are based on empirical evidence and knowledge of natural analogues. The possibility of a canister with an initial penetrating defect being emplaced in the repository has been considered as an incidental deviation in the analysis. Both likely and less likely lines of evolution, including the possibility of disruptive events, are identified and assessed. Account is taken of the natural evolution of the environment, chiefly driven by climatic evolution, which imposes external loads on the repository system, and also of internal loads, chiefly arising from the effects of excavation and

Figure 1: Overview of the key modelling activities for performance assessment (green boxes), which may consist of several different modelling studies. Yellow ovals present the main input and outcomes of the models. Closure consists of backfill and plugs in other tunnels than deposition tunnels, shafts and investigation holes.



emplacement of the spent nuclear fuel and the engineered barriers. Conditions that could lead to deviations from performance targets and target properties and, in particular, to release of radionuclides are identified, and the likelihood and effects of the deviations from the expected evolution estimated whenever possible. Feedback on further research and development work is also provided.

The performance of the repository system has been systematically analysed in three time windows: *i)* during the excavation and operational period up to closure estimated to last about 100 years; *ii)* up until 10 000 years after closure; *iii)* beyond 10 000 years up to one million years.

Excavation and operation up to closure of the disposal facility

Repository construction and operation cause changes in the host rock, including increased groundwater flow, which also affects the groundwater composition, changes in the stress field and potential rock damage around the underground openings and increase in temperature due to the heat generated by the spent fuel. Further, introduction of foreign materials and the presence of oxygen have an impact on the geochemistry and thereby on the performance of the engineered barrier system.

The increase in flow rates in the rock volume surrounding the repository is estimated to be approximately two orders of magnitude compared with pre-construction rates. After backfilling and closure, the flow rates return to near pre-excavation rates; however, a few deposition holes with flow rates and transport resistances outside the range defined by the target properties may remain. The average salinity around the repository remains similar to the pre-construction phase. However, the increased groundwater flow into the repository volume may lead to either more dilute or more saline conditions locally at repository depth. The disturbed conditions are related to the main hydrogeological zones and the ONKALO facility, not necessarily to the repository panels themselves. Moreover, the disturbed conditions are likely to last a limited time, in the order of tens of years, and thus the impact on the performance of the buffer and backfill is limited.

Calculations of temperature evolution show a maximum temperature at the canister surface of 95°C assuming an unsaturated buffer and 75°C for a saturated buffer. The maximum rock temperature at the deposition hole wall is about 65°C at around 50 years after emplacement. Thus, temperature in the buffer will remain below 100°C as required.

Excavation will cause a damaged zone (EDZ) to form, especially below the tunnel floors, although the damage is probably not continuous based on a dedicated study in the ONKALO facility. In addition, excavation and the heat produced by the spent nuclear fuel may cause spalling or other types of stress-induced rock damage around the excavated openings. The uncertainties concerning the properties of the EDZ and the rock damage around the deposition holes are taken into account in groundwater flow modelling.

Before full saturation, some buffer and backfill material may be lost through piping and erosion. Based on calculated inflows to deposition holes and tunnels, some limited buffer and backfill loss is expected. The average buffer and backfill density does remain high enough that the necessary low hydraulic conductivity and sufficient swelling pressure will be achieved as the buffer and backfill saturate. Thus, the performance targets of the buffer and backfill are fulfilled, even considering piping and erosion.

The consumption of oxygen in the backfill and buffer will be relatively rapid, due to its reaction with pyrite and other accessory minerals. Thus, anoxic, reducing conditions will be quickly established around the emplaced canisters. The maximum corrosion depth on the surface of the copper canisters due to atmospheric and initially trapped oxygen is expected to be less than 0.5 mm. Cementitious leachates from grouting of fractures, from grout used to stabilise rock bolts and from the plug in the deposition tunnel may locally affect the backfill. However, no cement is in direct contact with the buffer and the flux of cementitious leachates reaching the buffer is estimated to be of little significance.

Post-closure period during the next 10 000 years

Over the next 10 000 years, the climate is expected to remain essentially as today, i.e. a temperate climate with a boreal ecosystem. Groundwater flow and composition will recover from the disturbances caused by excavation, and will slowly evolve as a response to naturally occurring gradients. Key processes during this period will be water uptake, swelling and homogenisation of the clays in the buffer, backfill and seals, and the decline of the residual heat from the spent nuclear fuel.

Crustal uplift will continue, but at gradually lower rates, and higher hydraulic gradients will develop close to the shoreline. After 1 000 to 2 000 years, the shoreline will have retreated far enough that further retreat will not affect the flow rates in the repository volume. The heat from the spent nuclear fuel increases the flow rates at the repository depth by a factor of 2 to 3 compared with the natural state during the first hundreds of years. The heat tends to result in an upward driving force for the water, but when combined with the stronger natural downward forces, the flow remains mainly downwardly directed. Heat production declines to very low levels after the first few thousands of years, and the flow returns to its natural state.

The effect of the tunnel EDZ and the rock damage around the deposition holes (including thermally induced damage) on local flow rates around the deposition holes an on flow-related transport parameters has been modelled using a discrete fracture network approach (DFN flow model). The presence of the damaged zone increases the connectivity of fractures and flow around the deposition hole, but the effects on the natural fractures are limited, and flow rates in natural fractures and the transport resistances in the vicinity of the deposition holes are consistent with target properties for most deposition holes.

After closure of the repository, the disturbances caused by the repository construction cease and the salinity field at repository depth recovers, but at a much slower rate than the flow field. The natural salinity state is reached within hundreds of years. Groundwater composition also stabilises and the variation seen during the operational period diminishes. At repository depth, the pH remains close to 7.5 and reducing conditions prevail. In the longer term, salinity, chloride concentration and total charge equivalent of cations all decrease very slowly, due to the infiltration of meteoric water, but the concentrations remain consistent with the target properties over the time window in question.

Groundwater flowing into the repository leads to saturation and swelling of the buffer and backfill. Initial differences in the density and swelling pressure will be largely evened out (homogenisation). The time to reach full saturation in the buffer is calculated as a few tens to several thousands of years, depending on the local hydraulic conditions. Calculations show that a sufficiently high buffer density will be maintained in spite of any expansion of the buffer into the backfill.

Various gradients, including thermal gradients associated with heat generated by the spent nuclear fuel, will drive thermo-hydro-mechanical-chemical evolution and lead to limited geochemical changes in the buffer and backfill. After saturation and development of the full swelling capacity, the changes will be even lower, constrained by diffusive processes. The production of sulphide via microbial processes in the buffer will be minor, but cannot be ruled out in localised zones of low backfill density. Further, the already minor impact of cementitious leachates on the buffer and backfill is estimated to diminish, due to low concentrations of alkalis in the leachates.

Sulphide is the main copper corrosion agent in anaerobic conditions. Microbially produced sulphide in the buffer is negligible in this period; sulphide supply from the backfill is limited by the precipitation of iron sulphide and losses to the rock mass. Moreover, the sulphide has to diffuse through a thick layer of bentonite to reach the canister. Corrosion calculations coupled with groundwater flow modelling, and taking account of the possibility of early buffer erosion, show that the total corrosion depth will be negligible during the first 10 000 years. Moreover, the initially intact canisters will remain intact for all conceivable loads that could occur during the first 10 000 years and thus the spent nuclear fuel remains contained within the canister.

Evolution during repeated glacial cycles up to one million years

Over the longer term, major climatic changes are expected, including permafrost, glaciation and associated sea-level changes. These changes affect the isostatic load, rock stresses, and groundwater flow and composition, as well as the mechanical and thermal evolution of the EBS and host rock.

During the continued temperate climate up to 50 000 years AP, there is a slight increase in the groundwater flow rates in the upper part of the bedrock, due to surface environment changes. The flow rates at repository depth are not significantly affected. The continuing infiltration of meteoric water results in slowly decreasing salinity so that, towards the end of this period, a few canister positions may experience dilute conditions, at least if the chemical reactions in the overburden and with the fracture coating minerals and rock matrix minerals are not taken into account. Dilute conditions could give rise to chemical erosion of the buffer and backfill.

Groundwater flow and salinity have been modelled for two representative periods of permafrost development, during which permafrost reaches depths of about 80 m and 300 m. The effects of an ice sheet have also been modelled considering an ice margin staying over the Olkiluoto Island for 1 000 years, and a constantly retreating ice sheet. Under permafrost conditions, the hydraulic conductivity in the rock is reduced by several orders of magnitude and infiltration of the water from the surface is very low. As a result, the groundwater salinities remain at the level prevailing before the onset of the permafrost. During ice-sheet retreat, the flow rates through the repository volume depend on the location of the ice margin with respect to the repository. While the repository is still below the ice sheet but the ice margin is close, the flow rates are significantly increased (by a factor of 4 to 7) and directed downwards. As the ice passes the site, the main flow direction is upwards and flow rates reduce as the distance to the ice margin increases. Some canister locations might then experience higher flow rates and lower transport resistances than those specified in the target properties. Nevertheless, for most of the deposition holes, the host rock target properties related to groundwater flow are fulfilled during ice-sheet retreat.

Although there is no evidence that fresh meltwater ever reached repository depth at Olkiluoto during the last glacial cycle or during the previous ones, dilute conditions around some of the deposition holes during a future ice-sheet retreat phase might be possible and thus could lead to chemical erosion of the buffer and backfill. Other geochemical properties (pH, redox conditions, chloride concentration, total charge equivalent of cations sulphur and iron species) are all expected to remain consistent with the target properties throughout the period, including during ice-sheet retreat and melting. Oxygen will be consumed within short distances from the surface and will thus not reach the repository level.

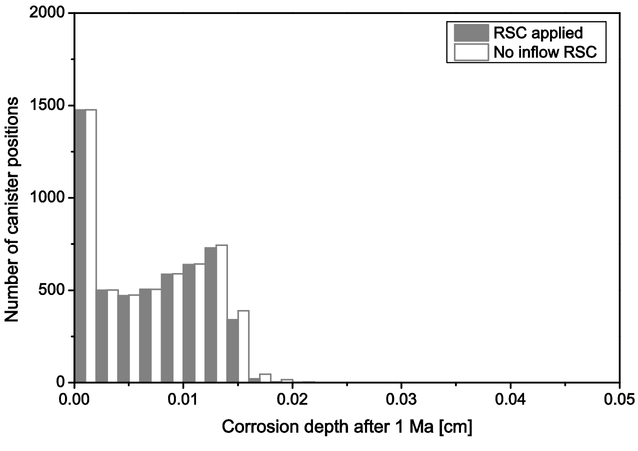
The possibility of a large earthquake leading to secondary shear movements on fractures intersecting deposition holes and to canister failure, especially during and after glacial retreat periods, cannot totally be excluded. The risk of canister failures due to secondary shear movements in the event of a large earthquake can be reduced by locating the deposition holes away from large deformation zones and by avoiding large fracture intersections in deposition holes, but it is estimated that a few tens of canisters may still be in positions such that they could potentially fail in such an event over a one million year time frame. On the other hand, the average annual probability of an earthquake large enough potentially to lead to canister failure due to secondary movements on fractures is estimated to be low, in the order of 10^{-7}. This is based on the frequency of occurrence of earthquakes in the Olkiluoto area and the fact that there are around five fault zones within and around the area of the repository that could host such an earthquake. Thus, during the first glacial cycle, there is little likelihood of canister failure due to rock shear, although the possibility of such failures cannot be discounted over a one million year time frame.

Freezing of the buffer or the deposition tunnel backfill is not a threat because, based on evidence from the past, permafrost will not reach the repository depth. In any case, the buffer and backfill would withstand the freeze/thaw cycles without damage to their safety functions. The evolution of porewater salinities in the buffer and backfill will follow those in the surrounding groundwaters, which will remain within the required target ranges, except perhaps for short times during ice-sheet retreat and melting period. Under these conditions, dilute groundwater conditions might cause some chemical erosion of buffer and backfill. With the reference assumptions on groundwater flow and evolving groundwater composition, one canister position is calculated to undergo buffer erosion during the first glacial cycle to an extent that advective conditions could arise. This calculation should be seen as illustrative, being based on only a single realisation of the DFN flow model. Taking a more cautious view on this and other uncertainties, buffer erosion might result in advective conditions in a few canister positions.

As at earlier times, sulphide is the main agent for corrosion of the copper canisters. Although groundwater data clearly indicate sulphide values below 1 mg/L, a pessimistic upper bound of 3 mg/L is adopted in corrosion calculations. The results show that, assuming the adopted value of 3 mg/l and that the buffer performs as designed, the overall corrosion depth will not exceed a few tenths of a millimetre even over one million years (Figure 2). Thus, no canister failures due to corrosion are expected. Furthermore, even if the buffer is affected by chemical erosion, few if any canister failures due to corrosion are expected during the first glacial cycle, as long as conditions otherwise correspond to the expected evolution. Using more cautious assumptions around three canister failures are calculated to occur within the first glacial cycle, and a few tens of failures in the million year time frame.

Figure 2: Number of canister positions as a function of corrosion depth over 1 Ma

With and without the rock suitability classification (RSC) inflow screening in the case of direct diffusion of sulphide across the buffer from a fracture intersecting the deposition hole with increased flow around the deposition hole due to rock damage; it is assumed that the groundwater sulphide concentration is 3 mg/L



Successive glacial cycles will impose similar loads as considered during the first glacial cycle. Thus, over the one million year assessment time frame:

- the potential for buffer erosion increases for deposition holes that experience dilute groundwater conditions during ice-sheet retreat;
- the number of deposition holes that suffer a shear displacement sufficient to cause canister failure could increase;

- the extent of canister corrosion in deposition holes that suffer buffer erosion could increase.

Conclusions

The analyses show that, under most conditions and lines of evolution, all performance requirements for the host rock and engineered barriers will be met. In this case, the copper canisters will remain intact and no radionuclide release will occur over at least one million years. Uncertainties in the initial state of the barriers and in the long-term evolution of the repository system, in particular during glaciations (Table 1), that may potentially lead to radionuclide releases are considered in various radionuclide release scenarios.

Table 1: Summary of deviations from performance targets and target properties as may occur and are relevant in each time window

Deviations	Up to closure of the disposal facility	Up to 10 000 years	During repeated glacial cycles
Possibility of an initial penetrating defect in one or a few canisters	Ö	Ö	Ö
Higher flow rate or lower transport resistance than the target values for a few deposition holes	Ö	Ö	Ö
Groundwater composition outside the target range for a short time during operation and soon after closure for a few deposition holes	Ö	Ö	–
Low density areas in the backfill where sulphate reduction to sulphide cannot be ruled out	–	Ö	Ö
Erosion of buffer in some deposition holes due to long-term infiltration of meteoric water or dilute glacial meltwater	–	–	Ö
Canister failure by corrosion due to unfavourable groundwater conditions and buffer erosion	–	–	Ö
Canister failure due to shear displacements in fractures during ice-sheet retreat	–	–	Ö

Some uncertainties still remain, but they do not affect conclusions regarding long-term safety. During the coming years uncertainties will be addressed through further research and technological development (RTD) activities for the operational license application, to either resolve them through a modified design or gather further data to better understand their long-term safety impact. The focus of research and development in the coming years is on:

- a better understanding of the processes affecting canister corrosion and erosion of buffer and backfill;
- rock conditions in potential volumes of rock for the repository and the application of rock suitability classification (RSC) criteria for the selection of repository panels, tunnels and deposition holes;
- demonstration of the implementation of the repository system components at full scale according to the technical design and quality performance requirements.

Further investigations of the properties of the rock in the repository area will reduce the probability of locating the canisters in unfavourable positions with respect to future loads. The processes affecting the performance of the engineered barriers will continue to be experimentally studied. Technical tests will be carried out to demonstrate that the repository can be implemented according to the assumptions made in the safety case.

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TURVA-2012: Formulation of radionuclide release scenarios

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Introduction

TURVA-2012 is Posiva's safety case in support of the Preliminary Safety Analysis Report (PSAR) and application for a construction licence for a repository for disposal of spent nuclear fuel at the Olkiluoto site in south-western Finland. This paper gives a summary of the scenarios and the methodology followed in formulating them as described in *TURVA-2012: Formulation of Radionuclide Release Scenarios* (Posiva, 2013). The scenarios are further analysed in *TURVA-2012: Assessment of Radionuclide Release Scenarios for the Repository System* and *TURVA-2012: Biosphere Assessment* (Posiva, 2012a, 2012b).

The formulation of scenarios takes into account the safety functions of the main barriers of the repository system and the uncertainties in the features, events, and processes (FEP) that may affect the entire disposal system (i.e. repository system plus the surface environment) from the emplacement of the first canister until the far future. In the report *TURVA-2012: Performance Assessment* (2012d), the performance of the engineered and natural barriers has been assessed against the loads expected during the evolution of the repository system and the site. Uncertainties have been identified and these are taken into account in the formulation of radionuclide release scenarios. The uncertainties in the FEP and evolution of the surface environment are taken into account in formulating the surface environment scenarios used ultimately in estimating radiation exposure.

Formulating radionuclide release scenarios for the repository system links the reports *Performance Assessment* and *Assessment of Radionuclide Release Scenarios for the Repository System*. The formulation of radionuclide release scenarios for the surface environment brings together biosphere description and the surface environment FEP and is the link to the assessment of the surface environment scenarios summarised in *TURVA-2012: Biosphere Assessment.*

Scenario formulation

To assess the impact of uncertainties in the evolution and performance of the repository system and the surface environment, a range of scenarios, each representing one or more possible time histories of conditions, or “lines of evolution”, are formulated and analysed.

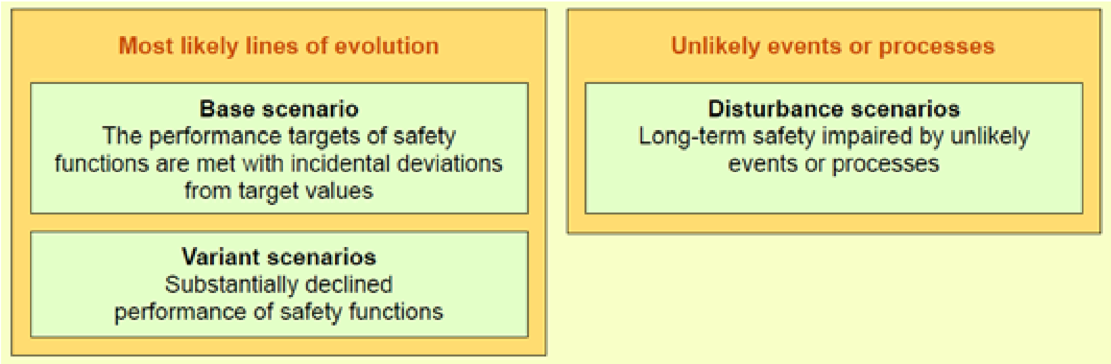
Consistent with Finnish regulatory and international guidance (Guide YVL D.5; IAEA, 2009, 2011, 2012), Posiva distinguishes between the expected evolution of the disposal system and unlikely events and processes. Account is also taken of the time window (or windows) in which releases of radionuclides might occur.

Guide YVL D.5 states: “Compliance with the requirements concerning long-term radiation safety, and the suitability of the disposal method and disposal site, shall be proven through a safety case that must analyse both expected evolution scenarios and unlikely events impairing long-term safety.”

The guide goes on to define three types of scenarios (Figure 1):

- *Base scenario:* The base scenario shall assume the performance targets for each safety function, taking account of incidental deviations from the target values.
- *Variant scenarios:* The influence of declined performance of a single safety function or, in case of coupling between safety functions, the combined effects of declined performance of more than one function shall be analysed by means of variant scenarios.
- *Disturbance scenarios:* Disturbance scenarios shall be constructed for the analysis of unlikely events impairing long-term safety.

Figure 1: Classification of scenarios in TURVA-2012, consistent with STUK’s Guide YVL D.5



The repository system is designed in such a way that each component of the engineered barrier system (EBS) should, for most likely lines of evolution and in the absence of incidental deviations, meet the safety functions and performance targets assigned to it, and the host rock should conform to its target properties. In this case, the copper-iron canisters remain intact for the whole assessment time frame and there is no release of radionuclides.

The performance assessment shows, however, that there are some plausible conditions and events (incidental deviations) that could lead to reduction of one or more safety functions, and thus may give rise to radionuclide releases. In addition, there are some very unlikely events and processes that could disrupt the repository, e.g. related to human intrusion and rock shear. These incidental deviations and unlikely events are systematically examined to define a set of scenarios that encompass the important combinations of initial conditions, natural evolution and disruptive events.

In the current and past assessments by Posiva, the scenario of a canister with an initial penetrating defect has been considered. This defect is assumed to be located in the canister weld as this is the least amenable component for quality checking once the canister has been loaded with the spent nuclear fuel. Although the likelihood that a canister with an initial undetected penetrating defect will be emplaced in the repository is low, it cannot currently be excluded and provides a useful base scenario for safety assessment (radionuclide release calculations) against which the efficiency of the other technical barriers and the host rock to limit the radionuclide releases can be tested and that also complies with the government decree GD 736/2008.

Thus, as indicated in Figure 1, the *base scenario* addresses the most likely lines of evolution (in which the performance targets and safety functions are met), but takes into account the possibility of one or a few canisters with initial undetected penetrating defects. The *variant scenarios* address situations that are considered reasonably likely and in which there may be reduced performance of one or more safety functions of the barriers. *Disturbance scenarios* address the lines of evolution that are considered unlikely but cannot be completely eliminated.

Methodology for scenario formulation

The repository system

Posiva's methodology for the formulation of radionuclide release scenarios relating to the repository system follows a top-down approach. The starting point for the repository system is presented in the *Design Basis* report (Posiva, 2012c), which accounts for the disposal concept, safety concept and the defined safety functions for the EBS and the host rock with their respective performance targets (PT) target properties (TP), all these considering the regulatory framework. The PT and TP are evaluated against the FEP affecting the system in the performance assessment and the lines of evolution resulting in deviations from the PT are brought further to the scenario formulation. In the scenario formulation the effects of single potentially detrimental FEP or combinations of FEP on the safety functions are considered systematically and also the effect of uncertainties within the expected lines of evolution. This systematic approach is designed to promote transparency and comprehensiveness. It can be summarised as follows:

FEP that could adversely affect one or more safety functions at a given time or place or under specific conditions within the repository are identified (i.e. FEP that are scenario drivers within the evolution of the repository system in time and space; see the *Performance Assessment* report).

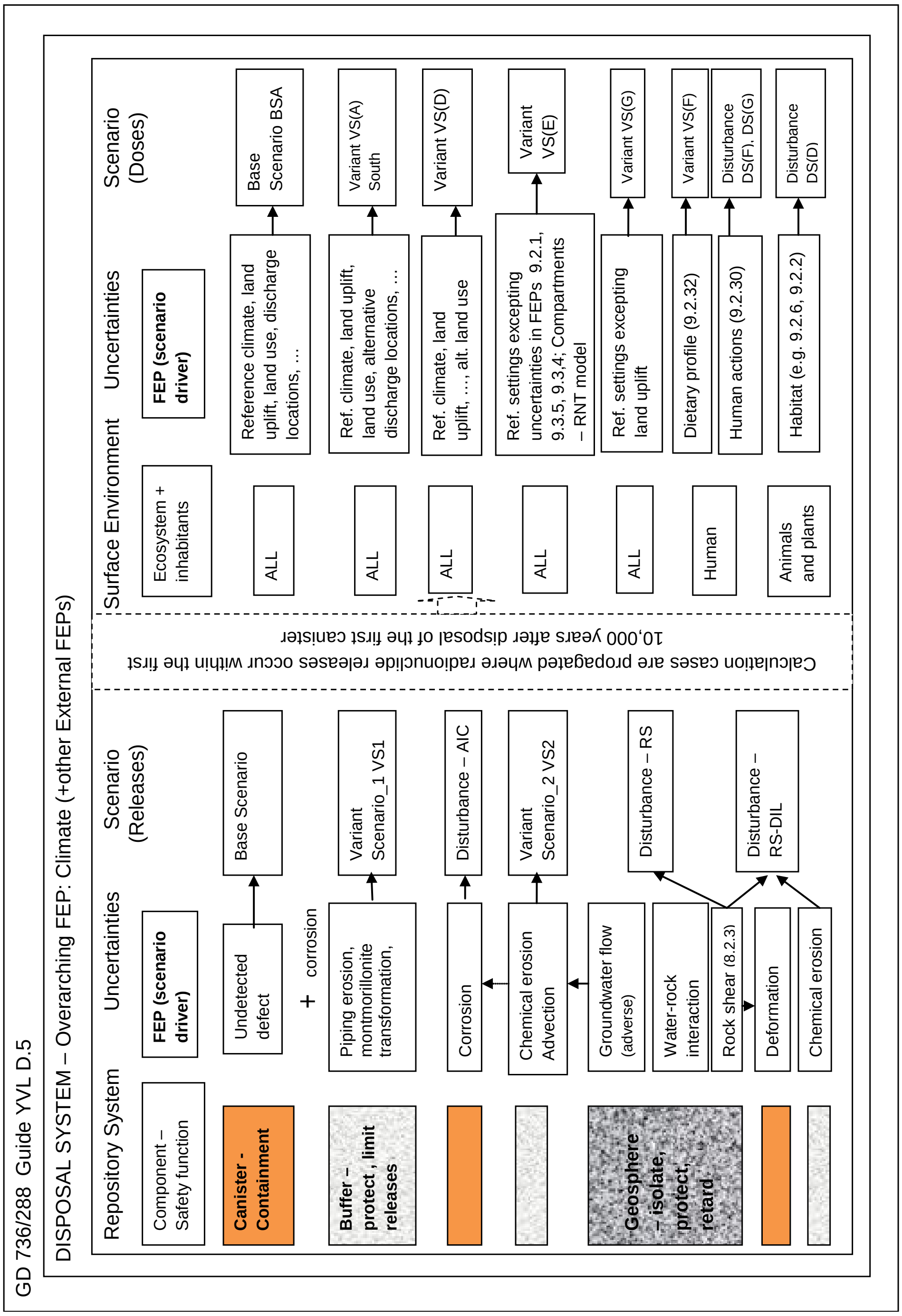
The effects of uncertainties in the expected evolution of the repository system (see the *Performance Assessment* report) are taken into account.

Thus, lines of evolution that describe the evolution of the repository system and ultimately lead to canister failure form the basis for the definition of radionuclide release scenarios. Each line of evolution is then classified using STUK's scenario terminology (Figure 1).

For each of the scenarios a set of calculation cases is defined to analyse the potential radiological impact. The calculation cases take into account uncertainties in model assumptions and data used to analyse the scenarios through variations in the models and parameter values.

The most important evolution-related FEP that may affect the safety functions of the repository system during its evolution, and thus affect also migration-related FEP, have been taken into account in analysing and describing the normal or expected evolution in the *Performance Assessment* report. Climate evolution is the overarching FEP affecting the whole disposal system. The outcome of the application of the methodology of scenario formulation is presented in Figure 2.

Figure 2: Outcome of the application of the methodology in scenario formulation



The surface environment

Formulation of scenarios for the surface environment must be consistent with the regulatory requirements, the methodology used in the formulation of scenarios for the repository system, and the current radiation protection systems for humans and the environment. Posiva's methodology for scenario formulation for the surface environment is somewhat different from that for the repository system, since the surface environment has no safety functions. Therefore, the scenario formulation for the surface environment is based on identifying FEP that may affect the evolution of the surface environment, fate of radionuclides in the surface environment and/or the potential radiation exposure of humans, plants and animals. The regulatory framework is also taken into account, mainly by coupling the scenario formulation to the dose constraints for humans. (GD 736/2008 Section 4: "Disposal of nuclear waste shall be planned so that radiation impacts arising as a consequence of expected evolution scenarios will not exceed the constraints", for which it is stated in Guide YVL D.5 paragraph 307: "In applying the dose constraints, such environmental changes needs to be considered that arise from changes in the ground level in relation to sea. The climate type as well as the human habits, nutritional needs and metabolism can be assumed to remain unchanged.")

Thus, Posiva's methodology for formulating surface environment scenarios can be summarised as follows:

- Constraints on the scenarios arising from the regulatory framework are identified.
- Key scenario drivers with respect to the evolution of the surface environment, fate of radionuclides in the surface environment and/or the radiation exposure of humans, plants and animals are identified. This work also comprises identifying FEP that affect the key drivers, either in isolation or combined, and could induce changes in a timeline of evolution.
- One or several lines of evolution are defined that describe the evolution of the surface environment, from which one or more scenarios are formulated. One credible line of evolution is identified and used to formulate the base scenario for the surface environment.
- Variant scenarios are formulated, mainly by considering reasonable deviations from the lines of evolution underpinning the base scenario. Variant scenarios can include additional scenario drivers with a potentially significant effect on the fate of radionuclides in the surface environment and/or the radiation exposure of humans, plants and animals.
- Disturbance scenarios are formulated, mainly by identifying unlikely FEP or mainly by considering unlikely deviations from the lines of evolution underpinning the base scenario. Disturbance scenarios can include additional scenario drivers with a potentially significant effect on the fate of radionuclides in the surface environment and/or the radiation exposure of humans, plants and animals.

Radionuclide release scenarios

Repository system scenarios

The base scenario

The canister (which provides the safety function of prolonged containment of the spent fuel) is the primary barrier, since radionuclide releases may only occur if the canister has failed. Possible canister failure modes are presence of an initial undetected defect corrosion, and rock shear. The base scenario postulates that one or a few defective canisters are emplaced in the repository. This is consistent with YVL Guide D.5, which states that the base scenario shall assume the performance targets for each safety function, taking account of incidental deviations from the target values.

Thus, in the base scenario reference case, an undetected penetrating defect in one canister is the incidental deviation that acts as the main driver, whereas the performance targets of all other repository components are assumed to hold, as is expected based on the performance assessment. The assumptions regarding the EBS and host rock (and surface environment) are shown in Table 1.

Table 1: Assumptions for the base scenario for radiological assessment

Surface environment	Climate evolution	Constant present climate for several millennia.
	Land use	Sparsely populated area. Crops, irrigation and livestock representative of present day. Forestry and peat land management according to present-day practice.
Bedrock	Rock mass	Rock suitability classification (RSC) criteria are applied successfully and target properties hold throughout the assessment time frame.
	Groundwater	Limited advection or inflows to repository level. Groundwater composition favourable to the EBS and target properties (according to RSC) for the groundwater composition holds throughout the assessment time frame.
Engineered barrier system (EBS)	Closure	Closure backfill and seals, including borehole seals are designed and emplaced according to requirements, and performance targets are fulfilled during throughout the assessment time frame.
	Deposition tunnel backfill	Deposition tunnel backfill and plugs are designed and emplaced according to requirements. The backfill performance targets are fulfilled throughout the assessment time frame.
	Buffer	The buffer is designed and emplaced according to requirements. The buffer performance targets are fulfilled throughout the assessment time frame.
	Canister	Canisters are manufactured and emplaced according to design. As an *incidental deviation* it is assumed that one or a few canisters are present with an initial undetected penetrating defect, whose size does not change in time.
Spent fuel + cladding		Very low dissolution rate; no specific requirements or safety functions.

Variant scenarios

The reduced performance of any single safety function(s) of any component other than the canister does not immediately give rise to canister failure and thus to radionuclide releases.

However, the reduced performance of the buffer may reduce canister lifetime and also subsequently affect radionuclide release and transport. The combined effect of the reduced performance of the canister and the buffer is assessed in two variant scenarios, where the loss of the safety function of the canister (initial penetrating defect or failure by corrosion) is combined with the reduced performance of the buffer.

Disturbance scenarios

In formulating disturbance scenarios, two main unlikely events are taken into account: one is the occurrence of a large earthquake capable of originating a rock shear that can breach the canister. The other is inadvertent human intrusion (treated in the *Biosphere Assessment*) for which an annual probability of occurrence is derived based on current habits and practices. FEP such as corrosion and deformation that are likely to occur, but only detrimentally affect safety functions if their rates are outside the expected range of possibilities, are taken into account in a scenario considering accelerated corrosion of the canister insert once the copper overpack has failed.

Surface environment scenarios

The base scenario for the surface environment and its main assumptions are as follows: The regulations explicitly state that the environmental changes due to sea-level changes relative to the land (i.e. allowing for land uplift) should be considered, and that the climate type as well as the human habits can be assumed to remain unchanged

(Guide YVL D.5, 307). Thus the current climate type in the region of the Olkiluoto site is assumed, as well as present-day demographic data and human habits, such as land use; site-specific or regional-specific information is preferred over national statistics.

Variant scenarios for the surface environment are based on alternative credible lines of evolution arising from reasonable variations of the FEP affecting the key drivers. Consideration has also been given to additional scenario drivers. The variant scenarios are listed in Table 2.

In the disturbance scenarios for the surface environment, unlikely lines of evolution that may have a potentially significant effect on the fate of radionuclides in the surface environment and/or the radiation exposure of humans, plants and animals are addressed. The identified disturbance scenarios are listed in Table 3.

Table 2: Variant scenarios identified for the surface environment, the driver the scenarios address and the uncertainties in the most important FEP affecting the drivers

Variant scenario		Scenario driver	FEP
VS-A	Discharge locations to the surface environment	Discharge locations	Defective canister location in the repository layout
VS-D	Land use (well)	Land use Human habits	Well (occurrence or not)
VS-E	Route of radionuclide transport	Element migration and accumulation	Alternative radionuclide transport routes in biosphere terrestrial and aquatic compartments affect a number of terrestrial and aquatic processes
VS-F	Exposure characteristics	Human habits	Uncertainties in dietary profile
VS-G	Combined scenario	Sea-level change (local) Land use	Agriculture and aquaculture – maximisation of cultivated areas

Table 3: Disturbance scenarios identified for the surface environment, the driver the scenarios address and the uncertainties in the most important FEP affecting the drivers

Disturbance scenario		Scenario driver	FEP
DS(D)	Exposure characteristics	Ecosystem occupancy	Habitats
DS(F)	Inadvertent human intrusion	Human actions	Human actions

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TURVA-2012: Assessment of radionuclide release scenarios for the repository system

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Introduction

TURVA-2012 is Posiva's safety case in support of the Preliminary Safety Analysis Report (PSAR) and application for a construction licence for a repository for disposal of spent nuclear fuel at the Olkiluoto site in south-western Finland. This paper gives a summary of the analyses of the radionuclide release scenarios formulated in a companion paper, *TURVA-2012: Formulation of Radionuclide Release Scenarios* (Marcos, 2014). The scenarios and the analyses take into account major uncertainties in the initial state of the barriers and possible paths for the evolution of the repository system[1] identified in a further paper: *TURVA-2012: Performance Assessment* (Hellä, 2014).

For each scenario, calculation cases are analysed to evaluate compliance of the proposed repository with regulatory requirements on radiological protection, as well as to illustrate the impact of specific uncertainties or combinations of uncertainties on the calculated results. Each case illustrates different possibilities for how the repository might evolve and perform over time, taking into account uncertainties in the models and parameter values used to represent radionuclide release, retention and transport and, for biosphere assessment calculation cases, radiation exposure.

The calculation cases each address a single, failed canister, where three possible modes of failure are considered:

1. In Posiva's terminology, the repository system refers to the system comprised of the spent nuclear fuel, EBS and the host rock.

- The presence of an initial defect in the copper overpack of the canister that penetrates the overpack completely (subsequent corrosion of the insert may then lead to an enlargement of the defect).
- Corrosion of the copper overpack, which occurs most rapidly in scenarios in which buffer density is reduced, e.g. by erosion.
- Shear movements on fractures intersecting the deposition holes.

However, the likelihood and consequences of more than one canister failure occurring during the assessment time fame are also considered, generally based on the findings from the single canister calculations.

Quantitative regulatory criteria regarding radiation protection are expressed in terms of nuclide-specific activity releases to the “living environment” (geo-bio fluxes) and annual doses (annual dose to the most exposed people and average annual doses to people). In addition, to address the qualitative regulatory requirements regarding protection of plants and animals, absorbed dose rates to representative organisms are also calculated. Thus, the safety indicators calculated in TURVA-2012 comprise geo-bio fluxes (expressed as normalised release rates[2] in the following sections), which are the main end-points considered in this paper, and the annual doses and absorbed dose rates to plants and animals evaluated in a biosphere assessment and reported in a further companion paper: *TURVA-2012: Biosphere Assessment* (Ikonen, 2014).

Main models and information flows

Figure 1 shows the main models and information flows used in the analysis of calculation cases. Models for the analysis of radionuclide release, retention and transport are shown in white boxes in the figure. Key supporting process models are shown as green boxes and system descriptions as light blue boxes.

Consistent with most safety analyses carried out internationally, the modelling of radionuclide release and transport in the repository system is carried out in two sequential steps: near-field release and transport modelling, and geosphere transport modelling, with the output from the former (near-field releases) providing input to the latter.

In the models used in the analysis of most calculation cases, radionuclides released from a failed canister are dissolved in water and conveyed in solution through the near field of the repository and through the geosphere towards the biosphere (gas- and colloid-mediated transport are also considered in some calculation cases). The most important radionuclide retention and transport processes included in near-field and geosphere transport models are advection, diffusion and sorption, and also solubility limitation in the near field.

Modelling results

Figure 2 shows the calculated peak normalised activity release rates from the geosphere to the surface environment for all calculation cases within the base, variant and disturbance scenarios.

The lowest peak normalised releases are for the reference case realisation (BS-RC) of the base scenario and sensitivity cases within the base scenario. In BS-RC, an incidental deviation is assumed whereby one canister with an initial penetrating defect of 1.0 mm

2. For each radionuclide, the normalised nuclide-specific release rate is a dimensionless quantity defined as the activity release rate of that nuclide divided by the respective regulatory nuclide-specific release constraint for that nuclide.

Figure 1: Models and information flows

Radionuclide release and transport models are shown in white boxes. System descriptions and understanding are shown in light blue boxes, key supporting models in green boxes and their principal outputs in dark blue ovals.

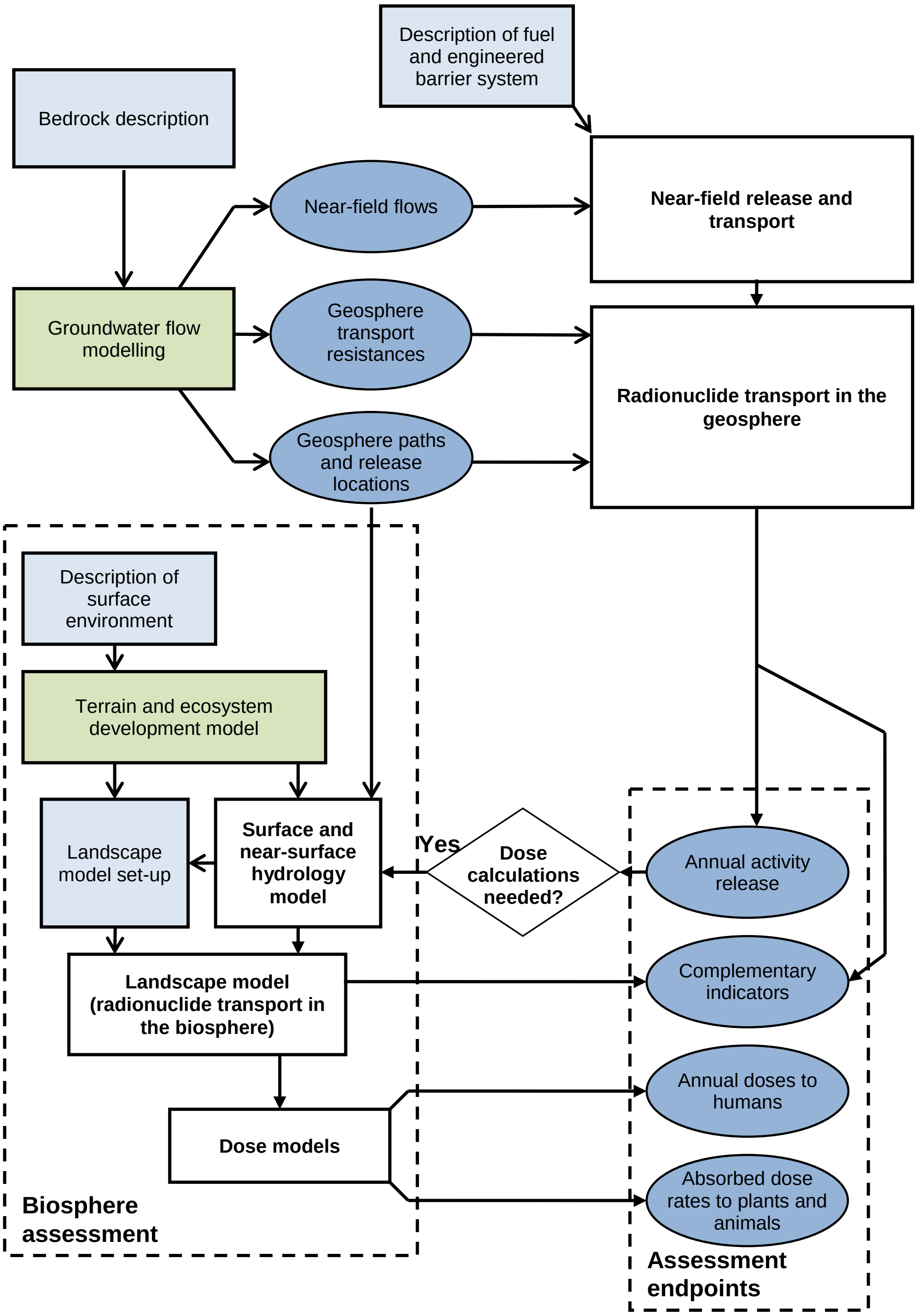
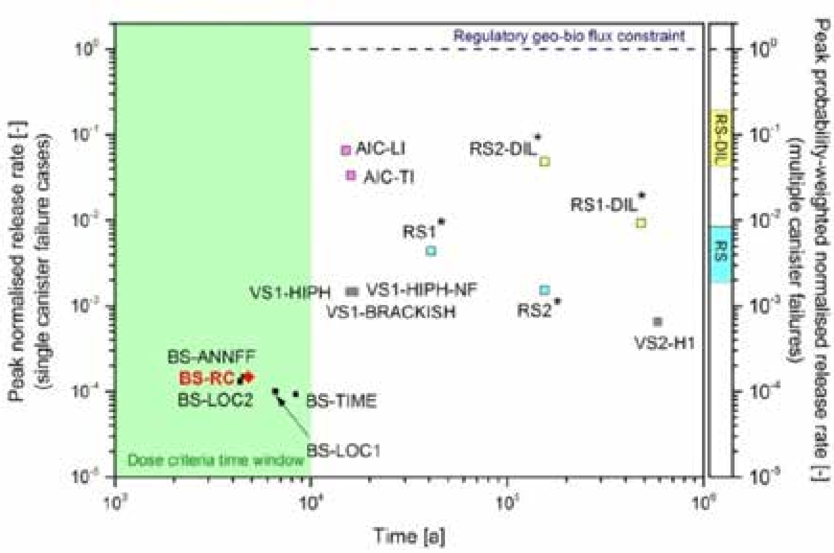


Figure 2: Peak normalised geosphere release rates for all calculation cases within the base, variant and disturbance scenarios, each assuming the failure of a single canister

Colours are used to group cases by scenario. * indicates that 1 000-year averaging is applied, in these cases. The right-hand subfigure shows ranges of values for the peak probability-weighted normalised release rates in the RS and RS-DIL scenarios. These ranges arise due to uncertainties in the numbers of canisters failing due to rock shear, as well as the timing of failure.



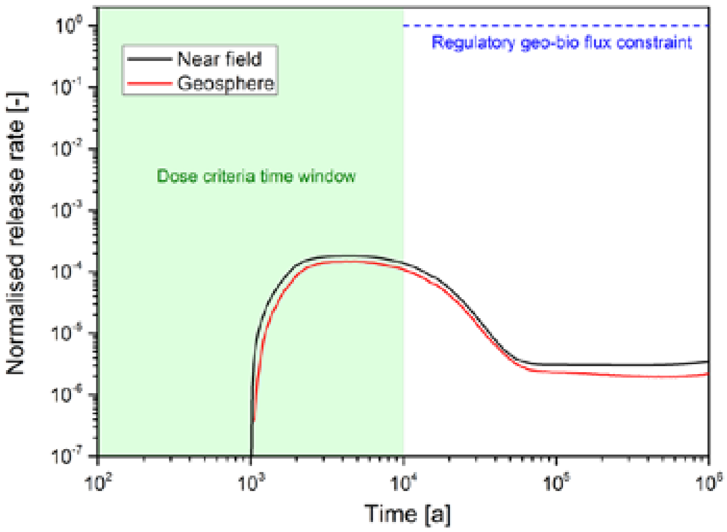
diameter is emplaced in the repository. All other canisters are assumed to comply with quality requirements. The single defective canister is cautiously assumed to be located in a deposition hole with relatively unfavourable hydrogeological characteristics. Except for the single defective canister, all other EBS performance requirements are assumed to be met and upheld during the evolution. Other cases within the base scenario consider alternative, cautiously selected positions for a canister with an initial penetrating defect within the repository and consequent different flow path characteristics, alternative near-field and geosphere speciation of radionuclides, and delayed establishment of the transport path through the canister defect.

Analysis of the reference case shows that the highest rate of radionuclide release is of ^{14}C, which peaks at around 4 500 years and then declines due to radioactive decay. Other, longer-lived radionuclides ^{36}Cl, ^{129}I and ^{135}Cs contribute at early times and dominate beyond a few tens of thousands of years. The dominant migration path is from the buffer into fractures intersecting the deposition hole; migration paths in the EDZ of the deposition tunnel or in the tunnel backfill are less important.

Figure 3 shows the near-field release and geosphere release rates for the base scenario reference case, normalised with respect to the radionuclide-specific constraint for the radioactive releases to the environment defined in STUK Guide YVL D.5. The release rates are summed over all calculated radionuclides and over the three release paths considered: from the buffer to the geosphere fractures intersecting the deposition hole; from the buffer to the EDZ of the deposition tunnel and thence into the geosphere; from the buffer to the tunnel backfill and thence to the geosphere. The figure indicates that during the dose criteria time window (up to 10 000 years) the normalised activity release is almost four orders of magnitude below the criterion of one as also given in Para. 313 of YVL D.5; beyond a few tens of thousands of years the normalised activity release rate decreases to between five and six orders of magnitude below one.

Figure 3: Evolution of the near-field and geosphere release rates for the base scenario reference case

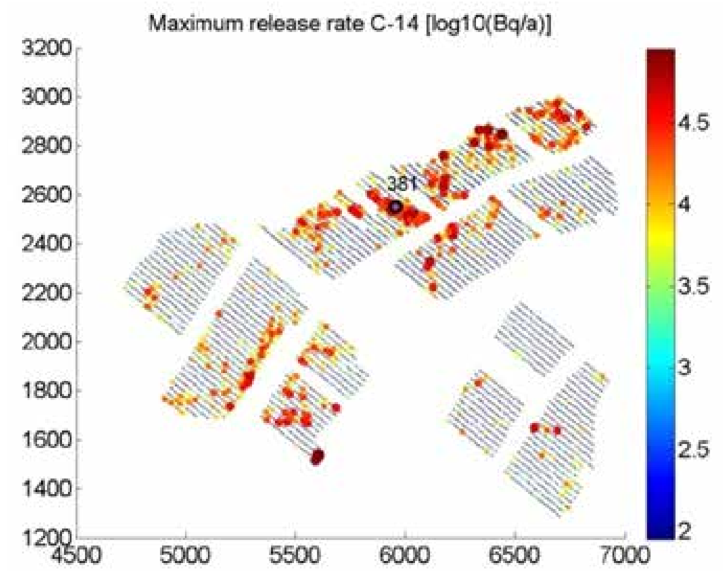
The release rate for each radionuclide is normalised with respect to the regulatory nuclide-specific constraints for radioactive releases to the environment. Regulatory geo-bio flux constraint denotes the constraint of 1 for the sum of the ratios between the nuclide-specific activity releases and the respective constraints given in YVL D.5.



The limited role of the geosphere in attenuating the peak release rate is related to the cautious assumption that the defective canister in the reference case is located in a deposition hole with relatively unfavourable hydrogeological characteristics. Figure 4 shows that, for most locations, the assumed canister defect would result in much lower ^{14}C peak release as most ^{14}C would decay during the transport in the geosphere.

Figure 4: Maximum geosphere release rate of ^{14}C as a function of location of the failed canister in the repository

The reference case location (381) is indicated by the black circle. The colour scales on the right-hand side indicate activity release rates in log10 (Bq a-1). Length scales are in metres.



There is a small possibility of there being more than one canister with an initial penetrating defect in the repository, and more than one of these canisters could be unfavourably located. However, based on an illustrative probability model for the reliability of the spent nuclear fuel disposal canister, the expectation value of release from multiple failed canisters randomly located in the repository is calculated to be significantly less than the release rate in the reference case, where a single failed canister at a cautiously selected location is postulated. Furthermore, the probability that the release maximum from multiple randomly placed defective canisters exceeds that of the reference case release maximum is low, estimated to be about 0.04%.

In Variant Scenario 1 (VS1) it is assumed that processes occurring at the buffer/rock interface lead to degradation of the outer part of the buffer and partial loss of its radionuclide retention capacity. Furthermore, there is an initial penetrating defect in one of the canisters. Enhanced transport of corrosive agents, such as sulphide, from the rock to the canister when the buffer is degraded may accelerate corrosion of the insert of this defective canister, as well as the overpack. It is assumed that the defect thus becomes enlarged over time due, for example, to volume expansion of the insert as it corrodes or to corrosion of the copper overpack. Results from cases designed to represent VS1 show that peak normalised release rates are about one order of magnitude higher than in the reference case, i.e. still almost three orders of magnitude below the regulatory requirement on the activity releases from the geosphere to the surface environment (regulatory geo-bio flux constraint). The peak is again dominated by ^{14}C but occurs later, at about 20 000 years, reflecting the influence of the progressively increasing diameter of the penetrating hole.

In Variant Scenario 2 (VS2), chemical erosion of the buffer takes place associated with ice-sheet retreat. Significant buffer erosion is considered unlikely, but cannot currently be excluded in at least some of the deposition holes. Eventually, advective conditions are established around the canisters in these deposition holes, leading to enhanced corrosion of the canister by sulphide, and eventually to canister failure (no initial penetrating defect is assumed but a thinner canister wall of 35 mm is adopted, which is the minimum thickness according to the design specifications). Taking into account results of the modelling of buffer erosion and sulphide corrosion from the performance assessment, canister failure is not expected to occur for at least several hundred thousand years. At these long times, the geosphere release rate is dominated by the non-sorbing and long-lived radionuclides, namely ^{129}I and ^{36}Cl. Modelled geosphere release rates also show periodic maxima, due to relatively rapid flushing of these radionuclides from the geosphere during periods of high flow associated with ice-sheet retreat. In the case representing the least favourable deposition position (VS2-H1), the peak normalised geosphere release rate for a single failed canister is more than three orders of magnitude below the geo-bio flux constraint. This low value indicates that the few canister failures that could potentially occur in the more likely lines of evolution (or even the few tens of canister failures calculated to occur based on highly pessimistic assumptions) could easily be tolerated without exceeding the regulatory constraint.

The rock shear (RS) scenario considers canister failure due to shear movements on fractures intersecting the deposition holes in the event of a large earthquake. Two cases have been analysed: RS1 and RS2, in which rock shear and canister failure are assumed to occur respectively at 40 000, i.e. during the present, temperate period, and at 155 000 years, during a period of ice-sheet retreat. The highest peak normalised release rates from the geosphere for a single failed canister are, in both cases, more than two orders of magnitude below the regulatory geo-bio flux constraint. This implies that more than one hundred canisters would have to fail simultaneously before the regulatory geo-bio flux constraint would be exceeded, even without taking into account the low probability that this event would actually happen. This exceeds the few tens of canisters estimated to be in critical positions that are vulnerable to failure in the event of a large earthquake.

In the scenario of rock shear followed by buffer erosion (RS-DIL), the buffer undergoes either immediate damage or longer-term erosion following the rock shear, due to the penetration of low-ionic strength water to repository depth. The peak release rates for RS-DIL cases are higher than for RS cases, but, nevertheless, the peak expectation value of the normalised release rate in the RS-DIL scenario, taking into account the uncertainty in the number canister failures and the timing of failure, is still around an order of magnitude below the regulatory limit.

The accelerated insert corrosion scenario (AIC) considers the possibility that an initial penetrating defect in a canister becomes enlarged over time due to faster than expected corrosion of the insert whereas the performance targets are fulfilled for all the other engineered barriers and the host rock is expected to meet the target properties during the evolution for the whole time window. More pessimistically than in VS1, the enlargement of the defect is assumed to occur instantaneously at 15 000 years leading to complete loss of transport resistance of the defect. The analysis of this scenario focuses on the significance of whether or not a transport path between the canister interior and the buffer exists prior to defect enlargement. Two cases have been considered: AIC-TI assumes no path exists before enlargement (i.e. the insert is water-tight and acts as a barrier) and AIC-LI includes such a path. In both cases, release rates increase rapidly at 15 000 years to peak shortly thereafter. The peak is somewhat lower in AIC-TI compared with AIC-LI. The largest normalised releases from the geosphere are in both cases at least one order of magnitude below the regulatory constraint.

To complement the deterministic analyses, Monte Carlo simulations and probabilistic sensitivity analysis (PSA) have been done. Both provide a rich source of understanding of the sensitivity of model outputs to variations in input parameter values, allowing the most important parameters and parameter combinations to be determined. It has been shown, for example, that ^{14}C, ^{36}Cl and ^{129}I control the normalised release rates for all cases where an initial penetrating defect in a canister is assumed, and that the properties of this defect (e.g. its size and evolution) are important in determining the peak release rates. A range of complementary indicators has also been evaluated. Their main roles are either to highlight the performance of certain components of the disposal system, or to provide an alternative line of argument for safety. In this context, radionuclide concentrations in the buffer and backfill have been shown to be comparable to examples of NORM, i.e. "naturally-occurring radioactive material", and radionuclide release rates to be comparable to naturally occurring activity fluxes in groundwater at the site.

Conclusions

The conclusions drawn from the assessment of radionuclide release scenarios are as follows.

Analysis of individual scenarios

Peak normalised releases for all calculation cases for the base scenario, variant scenarios and disturbance scenarios are below the regulatory geo-bio flux constraint, generally by more than an order of magnitude, even taking into account the possibility of multiple canister failures.

Combinations of scenarios

Plausible binary combinations of scenarios have also been considered. Many can be excluded from detailed analysis on qualitative grounds. Where appropriate, the release rate of two different scenarios has been summed, i.e. the base scenario in combination with corrosion failure following buffer erosion or in combination with rock shear; accelerated insert corrosion rate in combination with rock shear followed by buffer erosion;

corrosion failure following buffer erosion in combination with rock shear followed by buffer erosion; accelerated insert corrosion rate in combination with corrosion failure by buffer erosion. In each case, the combined release rate to the surface environment still does not exceed the regulatory constraint.

Sensitivities and uncertainties

Monte Carlo simulations, a probabilistic sensitivity analysis (PSA) and a number of deterministic complementary analyses have been performed so as to obtain a better understanding of the modelled system. The importance of the properties of any initial penetrating defect in the canister and its evolution over time has been highlighted in these analyses.

Complementary indicators

A range of complementary indicators has been evaluated. Radionuclide concentrations in the buffer and backfill have been shown to be comparable to examples of NORM and radionuclide release rates to be comparable to naturally occurring activity fluxes in groundwater at the site.

Quality control and assurance

Quality control and assurance measures have been adopted to ensure transparency and traceability of the calculations performed and hence to promote confidence in the analysis of the calculation cases. These include the verification of the assessment codes, validation of the codes for their intended applications, procedures to ensure codes are correctly applied, with an assessment database for the storage, checking and exchange of input data, intermediate results and final results; see also the companion paper in these proceedings, *TURVA-2012: Handling QA* (Snellman, 2014).

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TURVA-2012: Biosphere assessment

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Introduction

TURVA-2012 is Posiva's safety case in support of the Preliminary Safety Analysis Report (PSAR) and application for a construction licence for a repository for disposal of spent nuclear fuel at the Olkiluoto site in south-western Finland. Posiva's safety concept is based on long-term isolation and containment, which is achieved through a robust engineered barrier system (EBS) design and favourable geological conditions at the repository site. Nonetheless, if radionuclide releases were to occur that reach the surface environment, their transport and fate need to be modelled to assess the possible radiological impacts on humans, plants and animals.

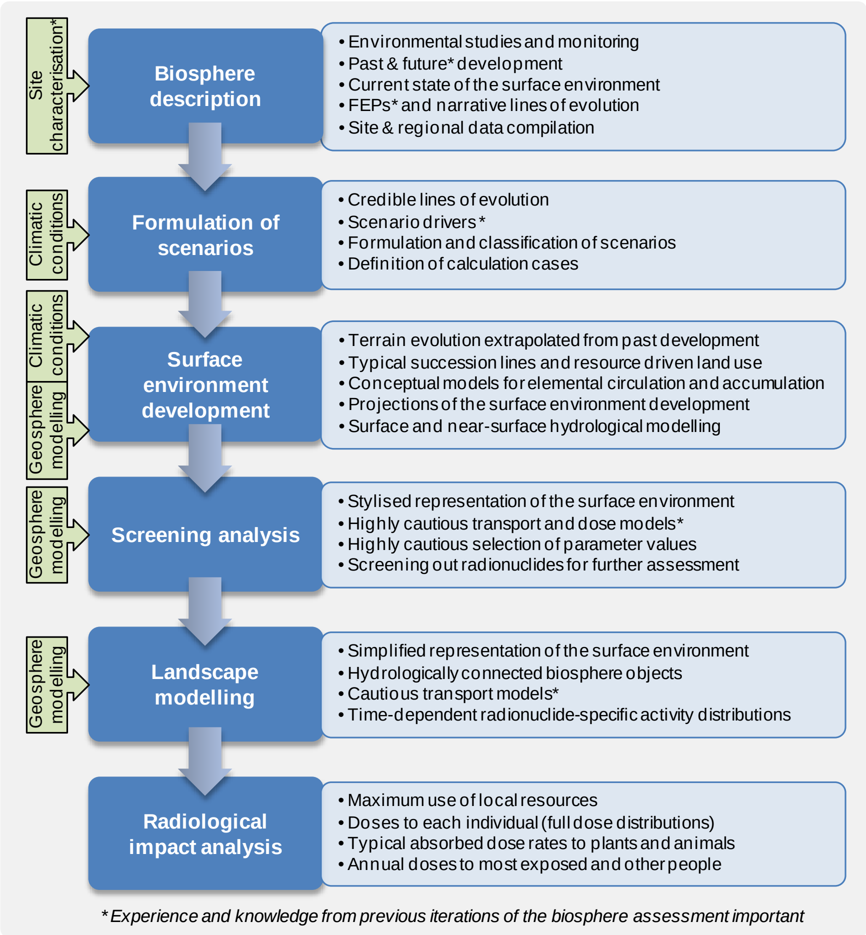
This paper gives a summary of the biosphere assessment of the Olkiluoto site (Posiva, 2012). The biosphere assessment describes the future, present and relevant past conditions at, and prevailing processes in, the surface environment at the Olkiluoto site. The transport and fate of radionuclides that may enter the surface environment as a result of hypothetical releases from the repository through the geosphere is modelled to finally assess the potential radiological impact on humans, plants and animals. The biosphere assessment follows the national regulatory requirements on the long-term safety of nuclear waste and aims to enable the assessment of compliance with the radiation dose constraints set out in the regulations. These requirements also contain guidelines on how to conduct the biosphere assessment, for example regarding which evolutionary features, events and processes (FEP) have to be accounted for and the exposure pathways that need to be considered in the dose calculations. However, when setting down the basic principles for the biosphere assessment and selecting the methodology to adopt when conducting the assessment, Posiva is not only aiming to be in line with the national requirements but also consistent with international recommendations, especially those made by the ICRP.

Method

Understanding the current surface environment and its past at the site and projecting its development into the future are two vital aims with the biosphere assessment, and these are tightly coupled. The key concept adopted is that the site is considered and modelled as a dynamic system. Alternative approaches could be to assume, despite the inherent dynamics of the surface environment, that the present conditions at the site would persist indefinitely, or to treat its future development by consideration of a set of time-independent snapshots of the possible futures. These approaches would be more appropriate for site generic studies, where detailed biosphere assessment would be too much a matter of conjecture, or for sites that are likely to remain broadly constant in characteristics over the period of interest for the assessment.

A fully dynamic approach is needed to fulfil the current assessment purposes, mainly since the surface environment at the Olkiluoto site will significantly change due to the post-glacial crustal rebound ("land uplift") on a time scale relevant to the demonstration of compliance with regulatory requirements and within the time frame associated with potential radionuclide releases from the geosphere. In particular, discharge areas for potential releases of repository radionuclides from the geosphere which are currently on the sea bottom will develop into terrestrial areas over a period of a few millennia. The current regulatory guidance (STUK Guide YVL D.5) reflects the potentially dynamic nature of repository sites and requires that this shall be accounted for in the assessment: "In applying the dose constraints, such environmental changes need to be considered that arise from changes in ground level in relation to sea." However, to better facilitate meaningful comparison to the experiences of the present, the guidance also allows that "the climate type...can be assumed to remain unchanged." The overall approach to how Posiva conducts the biosphere assessment and how it is implemented is a conceptual process containing six sub-processes (Figure 1).

Figure 1: The conceptual process of biosphere assessment, its sub-processes (blue boxes) and key features and products of each sub-process. The green boxes show the main inputs from activities external to the biosphere assessment in the TURVA-2012 safety case.



Scenarios and cases in the biosphere assessment

A base scenario has been formulated for the surface environment, centered on one credible line for the evolution of the surface environment, the fate of radionuclides in the surface environment and/or the radiation exposure of humans, plants and animals. Variant scenarios are formulated, mainly by considering reasonable deviations from the lines of evolution underpinning the base scenario. In addition, disturbance scenarios are formulated, mainly by identifying unlikely FEP or by considering unlikely deviations from the lines of evolution underpinning the base scenario. A set of biosphere calculation cases has been defined to quantitatively analyse the surface environment scenarios. A reference case, representing one model realisation of the base scenario is identified, as has a set of sensitivity and "What if?" cases to analyse variant and disturbance scenarios, respectively. The analysis of biosphere calculation cases is done stepwise in a modelling chain taking into account the connection between each biosphere assessment sub-process (see Figure 1). Twenty-one (21) cases are first defined and calculated within the terrain and ecosystem development modelling (TESM), which results in a series of projections of the development of the surface environment. It must be noted that only 2 of the 21 cases simulated have been propagated to subsequent models in the biosphere assessment process (the reason for not propagating all 21 cases is that most projections are considered to be similar to the reference case or bounded by other propagated simulation combining assumptions of reasonably maximised terrestrial proportion of the area with a reasonably maximal extent of cultivated crops). Biosphere calculation cases for the rest of the modelling chain are then identified for relevant scenarios (see Table 1) similar to the earlier stage; in each sub-process, the FEP to be addressed are identified and models and parameter values are selected accordingly. This is done independently for the surface- and near-surface hydrological model (SHYD), the landscape model (LSM), and the dose models used in the radiological impact analysis (RIA). It is also checked that the settings for each modelling step are consistent with both each other and with the scenario and the case to be analysed.

Table 1: Biosphere calculation cases under the base and variant scenarios

Calculation case	Calculation case in the sub-process modelling					Comments
	TESM	SHYD	LSM	RNT	RIA	
BSA-RC	REF	REF	REF	REF	REF	Reference case for the base scenario for the entire disposal system
VS(A)-SOUTH1	REF	REF	SOUTH	Southern_1	REF	Uncertainties in discharge locations to the biosphere
VS(A)-SOUTH2	REF	REF	SOUTH	Southern_2	REF	Uncertainties in discharge locations to the biosphere
VS(D)-WELL	REF	MORE_WELLS	REF	REF	REF	More wells than in the reference case
VS(D)-NO_WELL	REF	NO_WELLS	REF	REF	REF	No wells; irrigation and drinking water from surface waters
VS(F)-FINDIET	REF	REF	REF	REF	FINDIET	Dietary profile based on current average consumption statistics

The source term to the analysis of the biosphere calculation cases is the result of analysing repository calculation cases that give locations and rates of radionuclide releases within the dose assessment time window (i.e. up to 10 000 years after disposal[1]). Most of the repository calculation cases propagated to biosphere assessment are analysed with the biosphere reference case (BSA-RC) (see Table 2).

Table 2: Repository calculation cases propagated to the biosphere assessment and the biosphere calculation cases used in the analysis

Repository calculation case	Description of the repository calculation case	Biosphere calculation case/s used in the analysis (cf. Table 1)	Name of the resulting calculation case combination
BS-RC	Base scenario – reference case	All identified biosphere cases for the relevant discharge locations and human intrusion scenario cases	BSA-RC (cf. Table 1)
BS-LOC1	Sensitivity case assuming alternative canister location leading to releases to the surface environment south of present-day Olkiluoto Island	VS(A)-SOUTH1	VS(A)-SOUTH1 (cf. Table 1)
BS-LOC2	Sensitivity case assuming alternative canister location leading to releases to the surface environment south of present-day Olkiluoto Island	VS(A)-SOUTH2	VS(A)-SOUTH2 (cf. Table 1)
BS-ANNFF	Sensitivity case assuming alternative near-field and geosphere speciation	BSA-RC	BSA-ANNFF
BS-TIME	Sensitivity case assuming delayed establishment of transport path	BSA-RC	BSA-TIME
VS1-BRACKISH	Sensitivity case assuming reduced buffer thickness	BSA-RC	BSA-BRACKISH
VS1-HIPH	Sensitivity case assuming reduced buffer thickness and high-pH groundwater	BSA-RC	BSA-HIPH
VS1-HIPH_NF	Sensitivity case assuming reduced buffer thickness and high-pH (near-field)	BSA-RC	BSA-HIPH_NF
AIC-LI	"What if?" case assuming a high insert corrosion rate and copper overpack deformation; identical to BS-RC in the dose assessment time window	BSA-RC	BSA-AIC-LI

The outcome of the reference case in the TESM is utilised to construct two landscape models: one model for radionuclide releases from the discharge locations north of the present Olkiluoto Island, hence suitable for analysing the releases from the repository system reference case (BS-RC) and all other repository calculation cases that are based on

1. Chosen based on the regulatory requirements. It should be noted that within the regulations, the radionuclide releases from the geosphere to the biosphere in the longer term are constrained directly, instead of requiring an explicit dose assessment.

the same location of the failed canister, and one model for radionuclide releases from the alternative discharge locations south of the present Olkiluoto Island (the radionuclide releases in the cases BS-LOC1 and BS-LOC2). These two landscape models are denoted REF and SOUTH in Table 1. The biosphere objects (continuous and reasonably homogeneous segments of the area included in the surface environment development modelling) receiving the direct releases differ in the cases BS-LOC1 and BS-LOC2 both from each other and from the BSA-RC, which has implications for the radionuclide transport modelling part of the landscape modelling sub-process. This is reflected by two model variants for the model SOUTH (denoted Southern_1 and Southern_2 in the RNT column in Table 1). A reference case model is set up in the surface and near-surface hydrology (SHYD), denoted REF in Table 1. In addition to this, two SHYD cases are identified to analyse the variant scenario VS(D) addressing uncertainties in the number of wells at the site (denoted MORE_WELLS and NO_WELLS in Table 1).

In the terrain and ecosystem development line that maximises the land area for cultivated crops (Terr_maxAgri, Table 1), alternative sub-models and data are used for land uplift, climate characteristics, sedimentation in lakes and the extent of agricultural land. Also the water body connections, the surface and near-surface hydrology and the landscape model change accordingly compared to the reference case. The only models that are not affected by this alternative TESM projection are the dose models in the RIA.

In the BSA-RC, the radionuclide releases from the geosphere are assumed to enter the biosphere in the deepest layer of the overburden. To account for uncertainties in the configuration of the overburden and the release path within it, a case (RNT1) has been defined assuming that the geosphere releases enter the biosphere directly into the rooting zone in the terrestrial ecosystems and directly into the water column in the aquatic ecosystems.

In the radiological impact assessment (RIA), the reference case model is used for most of the biosphere calculation cases. Uncertainties in the dietary habits for humans are taken into account by implementing a model assuming, following the regulatory guidance, that the future generations would have the same preferences regarding consumption of various food groups as the present-day Finnish population.

Link between repository and biosphere calculation cases

Nine repository calculation cases give releases to the surface environment within the dose assessment time window. These cases are propagated to the biosphere assessment and analysed with biosphere calculation cases. These repository calculation cases and the type of biosphere calculation cases that are used to analyse them are summarised in Table 2. The guiding principle is that the reference case for the repository system (BS-RC) is analysed with all identified biosphere calculation cases, and all other repository calculation cases are analysed within the biosphere reference case (BSA-RC). An exception is that some repository calculation cases lead to geosphere release to the south of present-day Olkiluoto Island, so it is more suitable to use the characteristics of this area in the biosphere analysis.

Main results of the radiological impact analysis

The screening analysis (see Figure 1) performed on the radionuclide releases from the geosphere in the repository reference case BS-RC resulted in six radionuclides with a non-zero activity being screened out from further analysis. The five radionuclides propagated all the way through the biosphere modelling chain in BSA-RC are ^{14}C, ^{36}Cl, ^{129}I, ^{93}Mo and ^{108m}Ag.

The annual doses to representative persons within the most exposed group (E_{most_exp}) and among other exposed people (E_{other}) are presented in Figures 2 and 3 for BSA-RC. The dose maximum for E_{most_exp} is 2.0 10^{-7} mSv and occurs at about the calendar year 5000 and the corresponding dose maximum for E_{other} is 1.3 10^{-9} mSv and occurs at about year 4000. These results are about 6-7 orders of magnitude below the regulatory radiation dose constraints. ^{14}C dominates the annual doses, a direct effect of ^{14}C dominating radionuclide releases in the repository calculation case BS-RC during the dose assessment time window. The irregular shapes of the dose curves are mainly the effect of the dynamics in the landscape model and of calculating the dose by summing exposure-pathway-specific contributions from several biosphere objects. The dynamics in the development of biosphere objects, especially changes in their geometries, have a strong influence on the resulting activity concentrations in both environmental media in contaminated biosphere objects and in the foodstuffs the objects produce. For example, development of a lake in a terrestrial ecosystem may lead to a steep increase in the activity concentration in the shrinking water volume of the lake drying out.

Figure 2: The annual dose to a representative person within the most exposed group (E_{most_exp}) and the contributions from each radionuclide for the calculation case BSA-RC

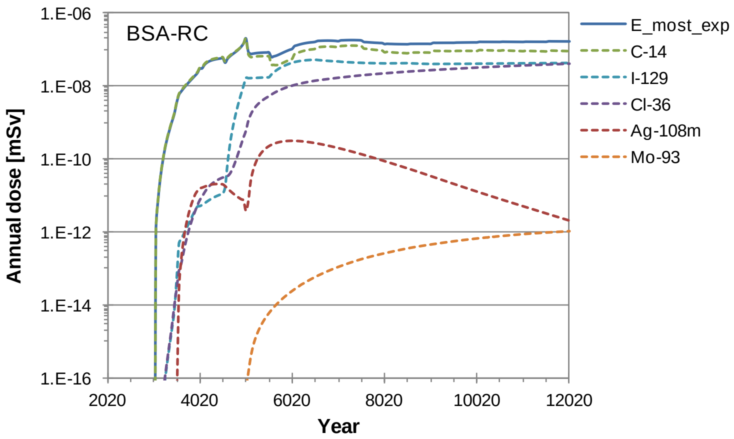
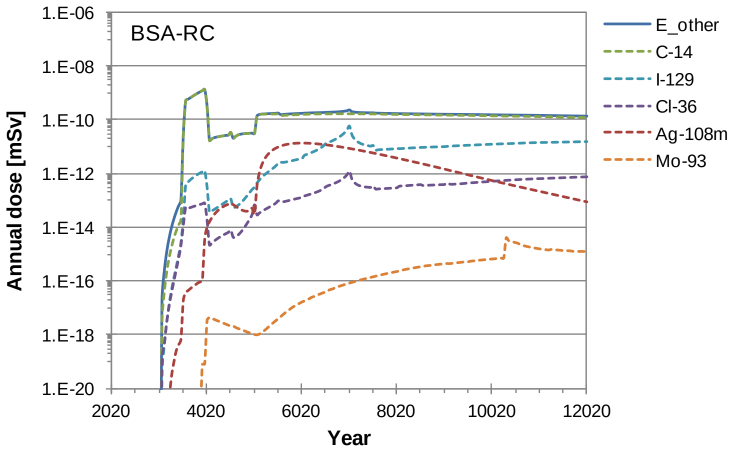


Figure 3: The annual dose to a representative person among other exposed people (E_{other}) and the contributions from each radionuclide for the calculation case BSA-RC



The (typical) absorbed dose rates for the most exposed representative plants and animals in freshwater, brackish, semi-aquatic and terrestrial environments are presented in Figure 4 for BSA-RC. The dose rate maximum over all organisms is 2.6 10^{-7} mGy/h, which is observed for pike in a freshwater environment.

The annual dose maxima to a representative person within the most exposed group (E_{most_exp}) and among other exposed people (E_{other}) are presented in Figure 5 for all calculation cases in Tables 1 and 2. Overall, the results remain below the regulatory constraints for humans and other biota.

Figure 4: Absorbed dose rate for plants and animals for the calculation case BSA-RC for the most exposed organisms in freshwater, brackish, semi-aquatic and terrestrial environments

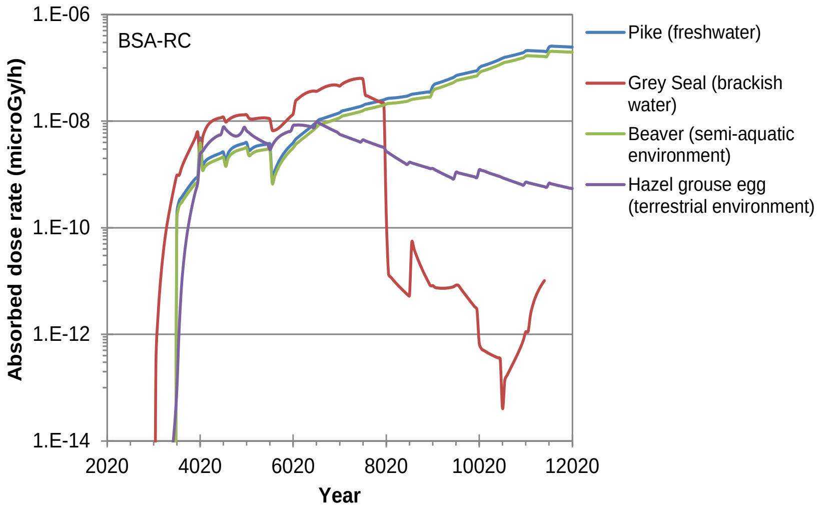
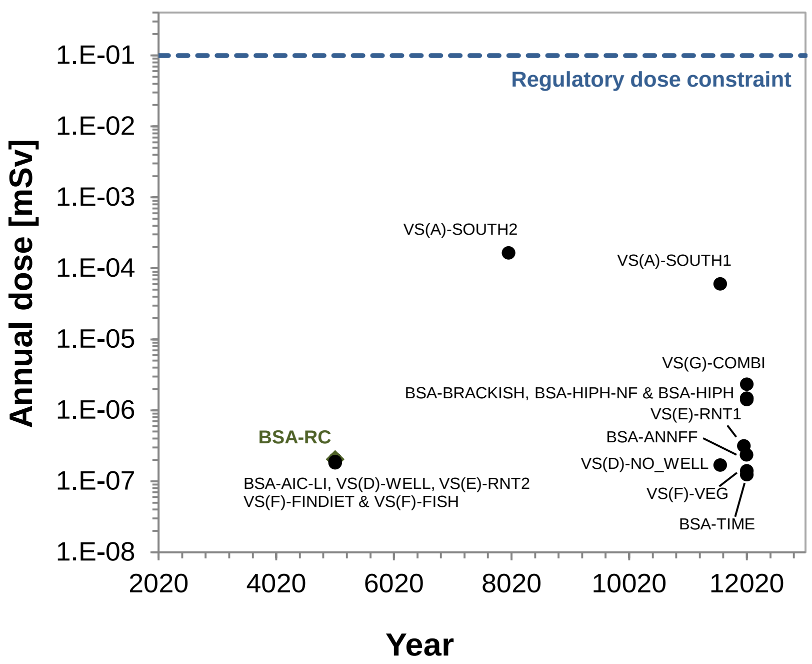


Figure 5: The annual dose maxima to a representative person within the most exposed group (E_{most_exp}) for the calculation cases listed in Tables 1 and 2



Reference

Posiva Oy (2012), *Safety Case for the Disposal of Spent Nuclear Fuel at Olkiluoto – Biosphere Assessment 2012*, POSIVA 2012-10, ISBN 978-951-652-191-9, Posiva Oy, Eurajoki.

Preliminary safety analysis of the Gorleben site

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Introduction

The safety requirements governing the final disposal of heat-generating radioactive waste in Germany were implemented by the Federal Ministry of Environment, Natural Conservation and Nuclear Safety (BMU) in 2010[1]. The Ministry considers as a fundamental objective the protection of man and the environment against the hazards of radioactive waste. Unreasonable burdens and obligation for future generations shall be avoided. The main safety principles are concentration and inclusion of radioactive and other pollutants in a containment-providing rock zone. Any release of radioactive nuclides may increase the risk for men and the environment only negligibly compared to natural radiation exposure. No intervention or maintenance work shall be necessary in the post-closure phase. Retrieval/recovery of the waste shall be possible up to 500 years after closure.

The Gorleben salt dome has been discussed since the 1970s as a possible repository site for heat-generating radioactive waste in Germany. The objective of the project preliminary safety analysis of the Gorleben site (VSG) was to assess if repository concepts at the Gorleben site or other sites with a comparable geology could comply with these requirements based on currently available knowledge (Fischer-Appelt, 2013; Bracke, 2013). In addition to this it was assessed if methodological approaches can be used for a future site selection procedure and which technological and conceptual considerations can be transferred to other geological situations.

The objective included the compilation and review of the available exploration data of the Gorleben site and on disposal in salt rock, the development of repository designs, and the identification of the needs for future R&D work and further site investigations.

Structure of the VSG

The VSG was composed of four main working topics:

1. Fundamentals: This topic included the description of the geological site and its future evolution over one million years; further, an inventory of the waste that could presumably be emplaced in a repository at the Gorleben site according to the current situation in Germany with its phase-out of nuclear energy (June 2011), and, finally, generation of a concept to accomplish radiological safety and to demonstrate its compliance with the safety requirements (BMU, 2010).
2. Based on these fundamentals, repository concepts were developed aiming toward operational safety, long-term safety and retrieval/recovering of the waste. Two emplacement variants, namely storage of spent fuels in drifts or in boreholes,

different types of canisters (POLLUX®, CASTOR®, BSK3R) and one optional variant emplacement for non-heat-generating waste were projected.

3. The analysis of the repository system was based on these concepts. The features, events and processes were compiled and described, then used to derive scenarios and to assess the probability of the evolution of the system. Geomechanical analyses investigated the integrity of the geological barrier (containment-providing rock zone) for 1 million years considering external and internal events and processes such as glaciation, decay heat or gas generation. Similarly, the seals for shafts and drifts were designed and analysed. The radiological consequences were analysed by numerical models for the transport of the liquid and gas phase (two-phase transport) in the long-term safety analysis.

4. The feasibility of the repository system as a containment system for radionuclides and the methodology to show compliance with the safety requirements were assessed. The uncertainties, which e.g. result from the incomplete geological exploration of the Gorleben site and which require additional R&D, were shown (Fischer-Appelt, 2013). The methodological approaches were discussed for their suitability to compare repository sites and their technical transferability to repository sites in other geological formations.

Geology

The Gorleben salt dome is 4 km wide and nearly 15 km long (Bornemann, 2011). It is composed of different salt rock types of the Zechstein (Upper Permian) series and extends to the Zechstein basin to a depth of more than 3 km. In the course of salt dome formation, the salt was moving several kilometres. During the uplift of the salt the initially plane-bedded strata of the Zechstein series were extensively folded. In the core of the salt dome the “Hauptsalz” sequence, which is characterised by a particularly high creep capacity, forms a homogeneous halite body with a volume of several cubic kilometres. The Hauptsalz contains gaseous and liquid hydrocarbons in separated zones of decimetre to metre dimensions. The overall hydrocarbon content is far below 0.01 weight-%. At the flanks, the salt dome consists of salt rocks with lower creep capacities. Brine reservoirs with fluid volumes in the range of litres to hundreds of cubic metres may exist in certain regions of this part of the salt dome. The water content of the Hauptsalz is below 0.02 weight-%. Interconnected pores do not exist in the salt rock outside of fluid-bearing or fractured areas, i.e. the salt rock is impermeable. The exploration of the Gorleben site as a potential site for a HLW repository started in 1979.

Based on this data a prognosis of the future evolution of the site was performed (Mrugalla, 2011). Geological and climatic features, events and processes were considered. The tectonic and volcanic activity, diapirism, subrosion, hydrology and climate were described and grouped into probable and less probable evolutions. Possible sequences of future glaciations were deduced from geological history.

Waste

The heat-generating radioactive waste will be composed of irradiated fuel elements from power reactors, vitrified reprocessing waste and irradiated fuel elements from research and prototype reactors. As an option, negligible heat-generating waste was also considered to be disposed to assess the feasibility of joint disposal in a separate area of the repository. A hypothetical amount and composition of this waste was assumed. It included depleted uranium tails from enrichment (about 35 000 m^3), graphite (about 1 000 m^3) and mixed waste (about 15 000 m^3) (Peiffer, 2011).

Safety concept

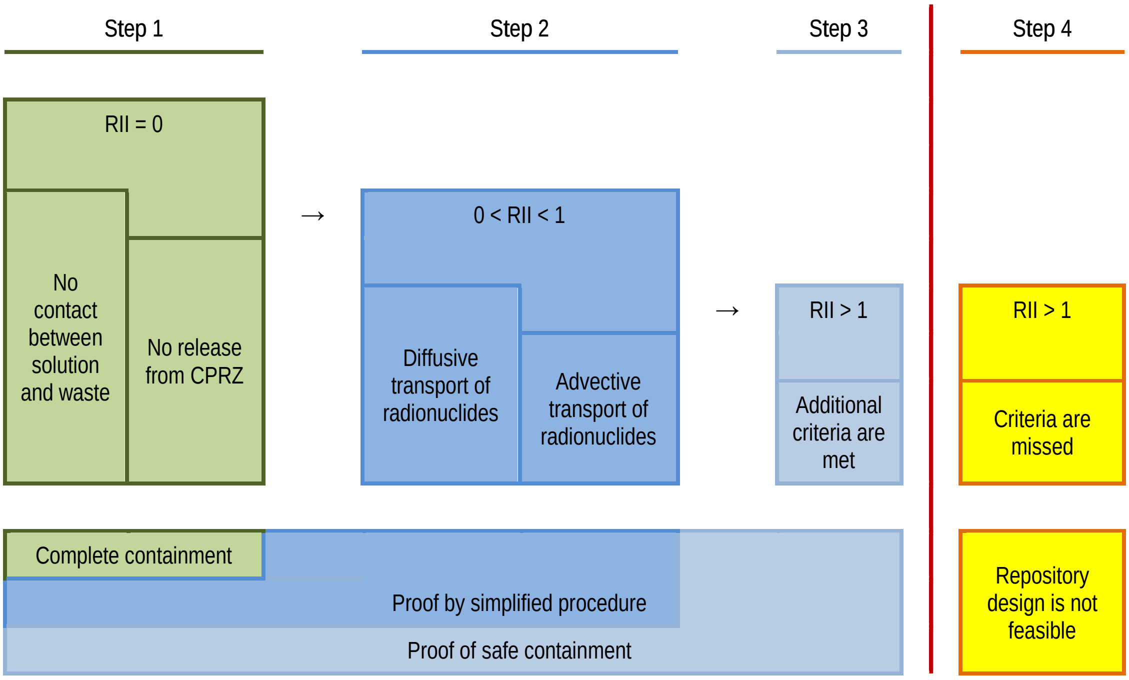
In due consideration of the German safety requirements (BMU, 2010) the safety concept for the project VSG was based on the following principles (Mönig, 2012):

- Radioactive waste must be contained in a containment-providing rock zone (CPRZ), i.e. a part of the host-rock enclosing the repository jointly with geotechnical barriers.
- Containment shall be effective immediately after closure.
- Containment must be provided by the repository system permanently and maintenance-free.
- Intrusion of brine to the waste forms shall be prevented or limited.

According to the safety requirements a site for disposal of heat-generating radioactive waste is only suitable if a sufficiently large containment-providing rock zone is available. Its integrity must be ensured for 1 million years and a robust, staggered, maintenance-free multi-barrier system from technical components (container, drift seal, shaft seal,...) must be developed, which prevents an unacceptable release of radionuclides over the short and long terms. This principle is also applicable if a barrier fails partially. Safe containment has to be demonstrated for probable and less probable evolutions of the site, while evolutions with very low probability (less than 1% over the demonstration period of 1 million years) need not to be considered. Criticality must be excluded in all phases of the repository development.

The evaluation of the safety of the system includes the assessment of probabilities and consequences. The proof of compliance with the safety requirements considers four steps and applies an indicator called the Radiological Insignificancy Index (RII) (Figure 1). The total release of radionuclides from the containment-providing rock zone and technical barriers is used in a generic model for radiation exposure. The RII is then calculated as a ratio to a radiation dose, which is considered insignificant (Mönig, 2012).

Figure 1: Radiological Insignificancy Index (RII)



The four steps considered in the RII are as follows:

- Complete containment is provided if there is no contact of the waste with solutions and no gaseous radionuclides are released from the containment-providing rock zone (RII = 0).
- Safe containment of radionuclides is achieved if the RII is greater than 0 and is less than 1. A simplified procedure is sufficient proof. The assessment distinguishes between a release by diffusion or advection.
- If the RII is greater than 1 additional criteria must be met. Additional criteria refer to the individual radiation dose and are related to the probability of the evolution of the system (scenarios). A more detailed procedure is required.
- If these additional criteria are unfulfilled the designed repository is not feasible. Safe containment cannot be provided by the repository concept. The design of the repository has to be changed and assessed once again. If all possible measures are optimised and safe containment still cannot be demonstrated, the site is not suitable.

Uncertainties and assumptions

There are some uncertainties for the Gorleben site, which cannot currently be further reduced or even eliminated due to the present status of knowledge, e.g.:

- The total lateral size of the salt dome is not yet known.
- The features of the salt rock are known only for the explored area.
- The extension of the Hauptsalz may not be large enough for all designed repository concepts, including the required safety distance to adjacent rock layers.

Therefore assumptions have been made which should be verified in the future. The main assumptions are:

- The lateral size of the salt dome is in accordance with the geological sketch of Bornemann (2011).
- The known features from the salt rock currently under exploration can be extrapolated to the entire area necessary for the whole repository.
- The extension of the Hauptsalz is sufficiently large for all designed repository concepts including the required safety distance to adjacent rock layers.

Geotechnical measures shall provide long- and short-term barriers. The long-term barrier is the backfill with salt grit. Its initial high porosity and permeability is reduced continuously by compaction. This is a time-dependent process and re-establishes the features of the undisturbed rock salt within the lifetime of the short-term barriers.

The short-term barriers are drift and shaft seals. These are composed by layers of different material providing diversity and redundancy. The failure of a drift or shaft seal is regarded as a less probable scenario.

Additional uncertainties concern data, parameters and models. These are dealt with in deterministic model calculations using bandwidths.

Repository concepts

The repository concepts for the Gorleben site are described for three emplacement variants (Bollingfehr, 2012):

- *Variant A*: As an option, non-heat-generating radioactive waste was emplaced in a separate area of the repository. This variant was combined with the following variants for spent fuel (Figure 2).
- *Variant B*: Emplacement of heat-generating radioactive waste (spent fuel and vitrified waste) in self-shielding waste containers (POLLUX[ã] casks) in horizontal drifts (Figure 2). As an alternative, the emplacement of heat-generating radioactive waste in transport and storage casks (CASTOR[ã]) in horizontal boreholes was considered, although this required an enhanced technical design for shafts and underground transportation.
- *Variant C*: Emplacement of heat-generating radioactive waste in multi-purpose conical overpacks (Figure 3) in deep vertical boreholes.

Figure 2: Repository design and layout (combination of Variants A and B)

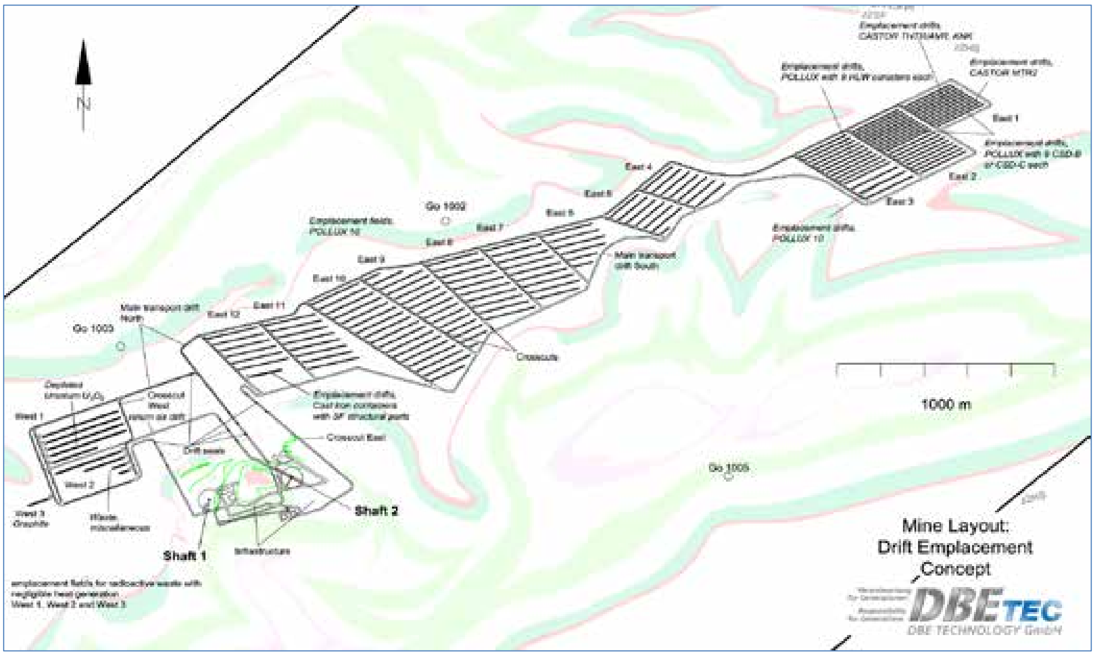
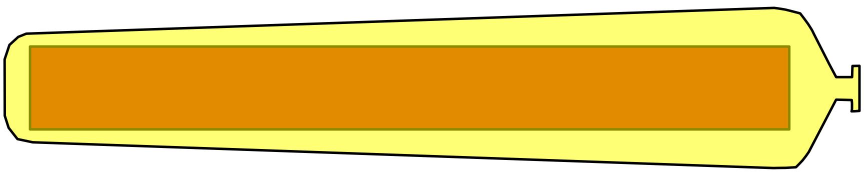


Figure 3: Conical overpack (Variant C)



The overall layout was optimised to minimise the size of the repository but also to comply with temperature criteria. The technical installation and casks/containers were selected to ensure manageability, radiation protection and operational safety. Disposal was planned using retreat working.

The technical solutions for retrieval of overpacks in Variant C were projected for the first time. A steel tubing (liner) of 300 m length was foreseen for disposal of the conical overpacks in boreholes. The void space in the liners was planned to be filled with dry quartz sand. The shape of the overpacks for spent fuel and flasks were designed conically to facilitate retrieval using vibration.

The sealing system for drifts uses sorel concrete and salt grit as backfill. Salt grit with a higher moist content is used as backfill for the main drifts to enhance compaction while for the emplacement fields salt grit with natural moist content is foreseen. The shaft seals are composed of bentonite, salt concrete and sorel concrete.

Scenario analysis

The site and the repository system will undergo exactly one evolution, which will be governed both by climatic and geological processes at the site and processes induced by the repository construction and the emplacement of heat-generating waste. This evolution cannot be predicted in all details.

A novel scenario development methodology was developed in the project VSG910. It aims at deriving one reference scenario for each repository design (horizontal drift/ borehole emplacement) and a number of differing alternative scenarios. At large, the scenarios shall comprehensively represent the range of possible repository system evolutions. The methodology allows the straightforward assignment of probability classes to the scenarios according to the regulatory framework (BMU, 2010). The individual scenarios are described by features, events and processes (FEP) that determine the future evolution of the final repository system at Gorleben. FEP may initiate or influence other FEP, be influenced by or result from other FEP (Wolf, 2013). These interdependencies were used to derive scenarios systematically. The reference scenarios were derived from probable FEP and basic assumptions. The alternative scenarios were generated from violation of assumptions, from less probable FEP and from probable FEP with less probable parameter values. The human intrusion scenario was reviewed separately (Beuth, 2013b).

System analysis

The system analysis addressed the following general questions:

- Will the integrity of the geological salt barrier remain intact under the expected loads, e.g. like heat generation or glaciation?
- Is there any flow of brine to the emplacement areas?
- Are radionuclides released from the containment providing rock zone?
- If so, what radiological consequences have to be expected?

Based on these scenarios an analysis of geomechanical and geotechnical integrity was performed (Kock, 2012; Müller-Hoeppe, 2012). A demonstration of integrity is required for probable scenarios (BMU, 2010). The integrity must be checked for less probable scenarios and their radiological consequences analysed. The final step was the assessment and synthesis of the results.

The dilatancy criterion and the fluid pressure criterion are the main criteria to assess the geomechanical integrity of the rock salt barrier. The dilatancy criterion specifies that no damage to the rock fabric (e.g. induced cracking or the interlinking of intercrystalline pore space) may occur in response to deviatoric stresses. The damage process is associated with dilatancy, i.e. an increase in volume caused by the development of micro-cracks and crack accumulations.

The fluid pressure criterion specifies that the smallest formation stress (considering compressive stresses as positive) in the barrier, plus any tensile strength which may be present, must be larger than the fluid pressure at a given depth. If this criterion is satisfied, fracturing of the host rock by fluid-pressure-driven penetration of fluids into the rock can be excluded.

The integrity of the salt barrier is only ensured if both criteria are satisfied in a sufficiently large zone around the underground workings of the repository. Linked flow paths from the water-bearing horizons in the overburden down to the emplacement zone, as well as release of hazardous substances from the repository itself (e.g. due to generation of a gas pressure) can then be excluded from a geomechanical point of view.

The mechanical and thermo-mechanical simulations carried out using a range of codes and material laws produced the following results and conclusions:

- The emplacement of heat-generating waste heats up the salt dome over a large volume, but the thermally-induced stresses and deformations do not generate any continuous migration paths.
- The highest thermo-mechanical stresses affecting the salt barrier occur within the first hundred years after sealing the geologic repository. Any loss of integrity of the barrier becomes even less likely in the subsequent time period. Mechanical damage caused by exceeding the dilatancy limit only affects the rock zones directly adjacent to the underground cavities within a few decimetres up to 3 metres and rock zones, which are restricted to the distant top salt zone. These rock zones at the salt top are of no importance with respect to the integrity of the salt barrier, which constitutes the containment-providing rock zone around the emplacement fields.
- The thermo-mechanical stresses calculated for the borehole emplacement design are higher than those calculated for the drift emplacement concept because the heat is released in a smaller and differently shaped volume.

The integrity of the geotechnical barriers (drift and shaft seal) was demonstrated by numerical calculations concerning geological, thermal and geochemical impacts during their lifetime and by providing redundant and varying types of sealing systems in combination.

Long-term safety assessment

For the analysis of the radiological consequences (Larue, 2013) radionuclide transport was modelled using a two-phase model, TOUGH2 (Pruess, 1999), and a one-phase model, MARNIE (Martens, 2002). The layout of the repository for the drift emplacement concept was transferred into a 3-D grid for TOUGH2 and a 1-D grid for MARNIE. This included simplification steps due to constraints of the codes.

Even a failure of a single seal did not result in advective flow of brine within or into the emplacement fields. Furthermore the salt grit is compacted in a relatively short time when applying conservatively selected parameters according to experimental data for modelling of the compaction process.

A conservative assumption was used for the final salt grit porosity after compaction. Using the 1-D grid no radionuclide transport by advection was detected in the liquid phase beyond the containment-providing rock zone (CPRZ). As a consequence any radionuclides in the liquid phase were transported by diffusion only. The release of radionuclides is higher through the eastern drift seal (Figure 2, close to cross-cut east) than through the western seal since it is closer to the disposal fields for spent fuel. The absolute value is a few Bq/a.

The radionuclide transport and release via the gas phase from the CPRZ is relevant up to some hundred years after closure in two-phase model calculations with TOUGH2. The RII at the drift seal exceeds in some model cases (BMU, 2010). The compaction of salt grit and metal corrosion with gas generation are driving forces on the transport and release of gaseous radionuclides (e.g. ^{14}C as methane from structural parts) through a drift seal. No ^{14}C was released via transport in the gas phase through a shaft seal.

Although many of the parameters for the MARNIE and TOUGH2 calculations were conservatively selected, they should be improved by future R&D work to improve the coverage of non-linear interactions. These are, e.g.:

- the compaction rate;
- the advective and diffusive transport parameters of radionuclides at low salt grit porosities;
- the diffusion coefficient in high compacted salt grit;
- the solubility limits for some radionuclides;
- the release of radionuclides into salt grit;
- the formation of gaseous radionuclides.

Synthesis of the project results

The safety concept, generated during the course of the project, was suitable to demonstrate its compatibility with the safety requirements. The generated design of the repository system was feasible and also complied with safety requirements. Nevertheless some assumptions were necessary. The assumptions refer to the status of geological exploration, the reliability of construction and some inherent uncertainties. If these assumptions are met, the designed repository system is assessed to be robust.

Optimising strategies regarding the repository design are conceivable. A repository layout such as placing the structural components farther away from the drift seals would likely result in a lower ^{14}C flow through the drift seals. Furthermore, implementing a void volume as a sink (e.g. an infrastructure area backfilled with gravel) might hinder any gas flow through the shaft seals. Further, the use of gas-tight casks for the structural components (like POLLUX®) could confine volatile radionuclides for decades or centuries.

Some conclusions were:

- The possible release of gaseous radionuclides, the two-phase flow processes and the subsequent model for radiation exposure require additional R&D.
- The containment-providing rock zone can be minimised by an iterative process but has to be preliminarily assessed as an initial guess.
- The handling of combinations of less probable but interdependent FEP has to be improved.
- The mobilisation of other pollutants and the heating of groundwater should be observed and requirements should be defined.
- The complete, though preliminary, safety analysis for the Gorleben site identified important tasks for research and development. This would not have been possible with a generic safety analysis.
- A safety analysis should be repeated at selected time intervals.
- Indications were given that a repository in salt rock is feasible.

Applicability to a site selection process

During the course of the project the request emerged in politics to establish a transparent, stepwise site selection process in Germany. Alternative sites should be identified and explored in addition to the Gorleben site. A bill was introduced for a site selection process and adopted in July 2013. This bill foresees a development of site selection criteria and standards for site comparison, which should be developed for different steps of the process. Safety analyses are foreseen to evaluate the results of surface and sub-surface explorations of possible sites.

The results of the VSG are valid for a repository site in salt rock. The methodology can be transferred to sites in other geological formations but requires a concrete concept for disposal and closure to perform a safety assessment and comparison of sites.

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Used fuel repository post-closure safety assessment in crystalline rock

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Abstract

The Nuclear Waste Management Organization (NWMO) is responsible for the implementation of Adaptive Phased Management (APM), the federally-approved plan for safe long-term management of Canada's used nuclear fuel. Under the APM plan, used nuclear fuel will ultimately be placed within a deep geological repository in a suitable rock formation.

The repository and its surroundings comprise a system that is designed to protect people and the environment through multiple barriers. These barriers include ceramic used fuel, long-lived corrosion-resistant containers, engineered sealing materials and the surrounding geosphere.

A site selection process is currently under way to identify a safe site in an informed and willing host community. The process of site selection will take several years. As potentially suitable sites are identified in interested communities, detailed field studies and geoscientific site characterisation activities will be conducted to assess whether the multi-barrier repository concept could be safely implemented to meet rigorous regulatory requirements.

At this early stage in the process, before specific sites have been identified for examination, it is useful to conduct generic studies to illustrate the long-term performance and safety of the multi-barrier repository system within various geological settings.

This paper summarises an illustrative case study of the current multi-barrier design and post-closure safety of a deep geological repository in a hypothetical crystalline Canadian Shield setting (NWMO, 2012). The purpose of this case study is to present a post-closure safety assessment methodology to illustrate how Canadian Nuclear Safety Commission (CNSC) expectations, documented in CNSC Guide G-320, Assessing the Long Term Safety of Radioactive Waste Management, are satisfied (CNSC, 2006). The approach is also consistent with international recommendations for the preparation of a safety case (IAEA, 2012), but this study specifically focusses on the post-closure assessment aspect and is not a full safety case. In the case of a licence application for a candidate site, a full safety case would be prepared to describe the site-specific geosphere, the repository design and operational and post-closure safety in support of a repository preliminary safety report and environmental assessment.

Introduction

The purpose of the post-closure safety assessment is to determine potential effects of the repository on the health and safety of persons and the environment. Results are compared against acceptance criteria established for the protection of persons and the environment from potential radiological and non-radiological hazards.

The approach is consistent with that outlined in CNSC Regulatory Guide G-320 *Assessing the Long Term Safety of Radioactive Waste Management*.

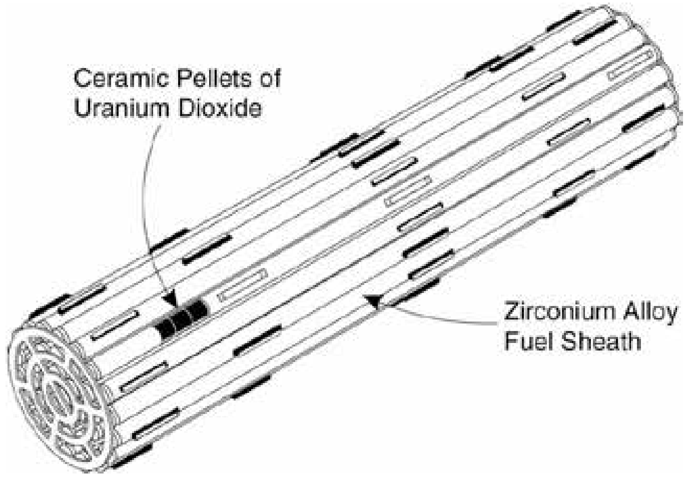
The assessment time frame extends from repository sealing and closure until the time at which the maximum impact is predicted, with a one million year baseline adopted based on the time needed for the used fuel radioactivity to decay to essentially the same level as that in an equivalent amount of natural uranium.

Waste form and design concept

Waste form

The reference waste form (see Figure 1) is a standard CANDU®[1] 37-element fuel bundle with a burn-up of 220 MWh/kgU and an average fuel power during operation of 455 kW. An un-irradiated fuel bundle contains about 20 kg of natural uranium in the form of ceramic pellets. Each bundle is about 0.5 m in length.

Figure 1: Waste form



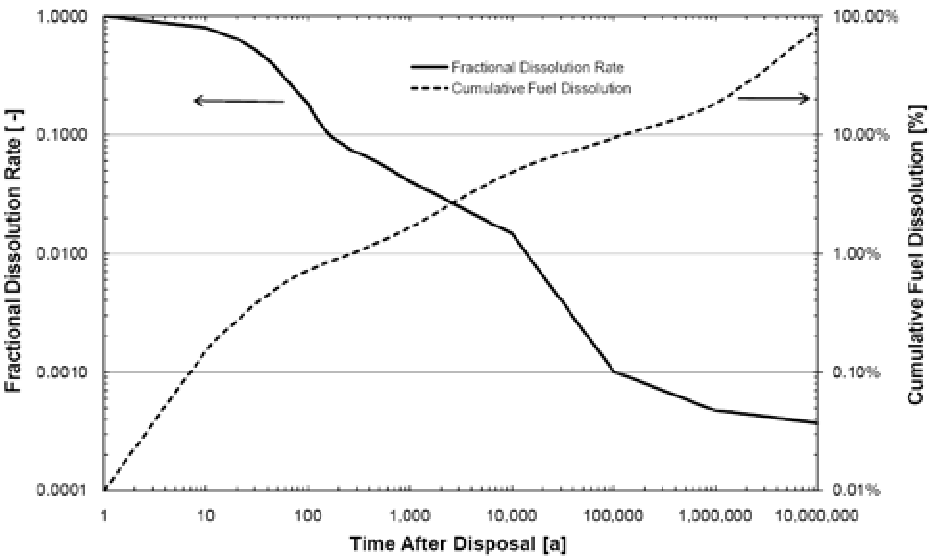
The used ceramic fuel matrix is durable and dissolves slowly in water. The most important factor governing the rate of dissolution is the electrochemical potential of the groundwater saturating the repository. The groundwater is anticipated to maintain a reducing environment in and around the container, with any residual oxygen available at the time of closure consumed by reactions with the enclosing engineered sealing materials and/or the copper and steel container itself.

Conditions at the used fuel surface are likely to remain oxidising for a long time due to radiolysis of groundwater that contacts the fuel should the container and fuel sheath fail. Radiolysis would be caused by alpha, beta and gamma radiation from the used fuel, at rates that decrease with time. Note that although used fuel dissolution experiments

1. CANDU® is a registered trademark of Atomic Energy of Canada Limited.

indicate that the dissolution rate decreases by several orders of magnitude in the presence of significant hydrogen gas as would be produced by corrosion of the steel container (Shoesmith, 2008), this effect has been conservatively ignored. Figure 2 illustrates the fuel dissolution as a function of time.

Figure 2: Fuel dissolution



Notes: Relative dissolution rate is the ratio of the time-dependent fuel dissolution rate to the maximum fuel dissolution rate. The maximum dissolution rate is 3.12 ´ 10^{-3} [mol/m^2/a] where the area is the surface area of the fuel in contact with water. A contact area of 1 570 m^2 per container is used in this study, which assumes the fuel is highly fragmented. The maximum dissolution rate is therefore 4.9 mol/a.

Design concept

In this illustrative case study, the reference design concept is similar to the KBS-3 repository method, with modifications to the container and internals as required to accommodate 360 used fuel bundles. The container is illustrated in Figure 3.

Figure 3: Used fuel container

The container design consists of a copper outer vessel, or shell, that encloses a steel inner vessel. The outer copper shell provides effective resistance to container corrosion under deep geological conditions, while the inner steel vessel provides strength for the container to withstand expected hydraulic and mechanical loads, including earthquakes and glaciation. Figures 4 and 5 illustrate the layout of the repository and placement rooms.

Figure 4: Underground layout

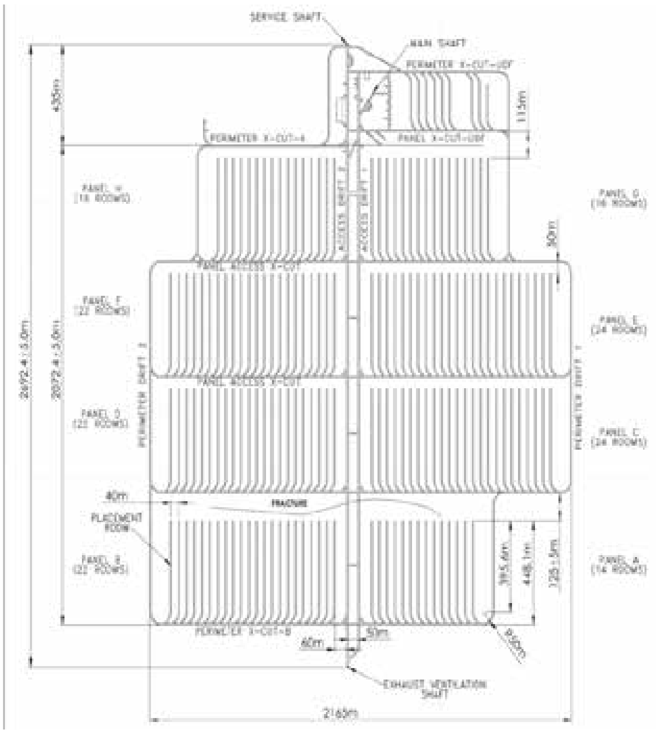
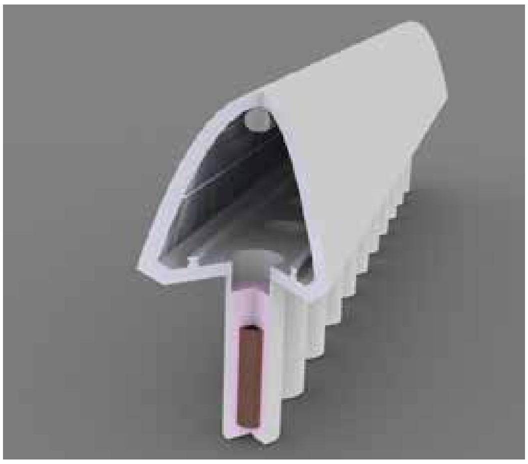


Figure 5: Placement room



The repository contains a network of placement rooms with in-floor boreholes for the base case inventory of 4.6 million used fuel bundles encapsulated in 12 800 used fuel containers. Each borehole in the floor along the placement room centreline has a used fuel container surrounded by a clay-based sealing material of highly-compacted bentonite buffer disks, rings and gap-fill pellets.

The remaining excavated spaces in the placement rooms are filled with engineered sealing materials made from mixtures of clay, sand and rock to minimise the movement of water. Placement rooms are sealed with bulkheads of low-heat, high-performance concrete. All tunnels and shafts are filled with similar engineered sealing materials, isolating the repository from the biosphere.

While not incorporated in this illustrative case study, it should be noted that the NWMO has a significant ongoing engineering design programme, the purpose of which is to optimise the container and repository design. Smaller containers with thinner copper coatings and different repository layouts are being considered.

Geosphere

The long-term safety and performance of a used fuel repository will rely, in part, on the surrounding geologic setting. The geosphere will provide a geomechanically and geochemically stable environment. Geomechanical stability enables safe excavation and placement of the containers and engineered barrier system (EBS), and isolates the containers from a wide range of future human and natural events. A stable geochemical and hydrogeologic environment supports container durability and minimises radionuclide mobility. The ability of the geosphere to support these attributes will be dependent on site-specific conditions.

In crystalline rock, fractures are the main pathway for water movement and mass transport. The hydrogeology of the setting can be generally characterised by a flow system decoupled into shallow and deep groundwater zones. In the shallow groundwater zone, groundwater flow is driven by topographic gradients through the more permeable rock mass. Advection is the dominant transport mechanism. The deep groundwater zone is stagnant and more isolated. Fractures are less common and less likely to be interconnected. The groundwater in this zone is old, slow-moving and chemically distinct. Diffusion would be the dominant mechanism for contaminant transport until the contaminant plume intersects a transmissive fracture.

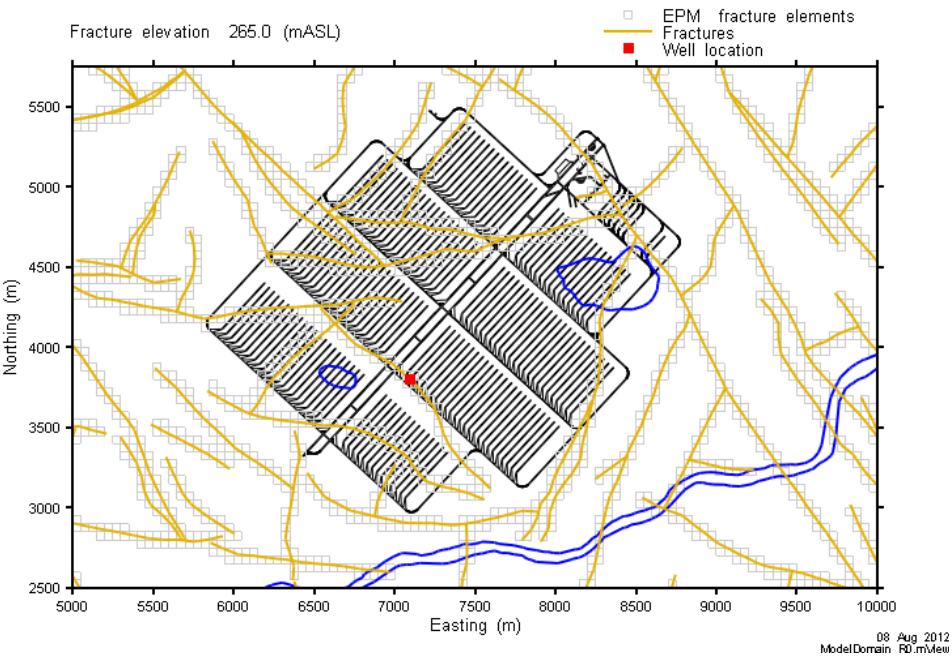
A hypothetical geosphere has been derived, in part, from historic experience gained in the Canadian Nuclear Fuel Waste Management Program. It was developed for the purpose of this illustrative case study while the NWMO proceeds with the APM siting process and selection of a preferred site in an informed and willing host community. While the hypothetical site represents one example of a possible crystalline rock setting, a range of characteristics are described for alternative settings that are considered in the safety assessment to illustrate an approach to assess long-term safety and the functionality of various barrier systems.

A set of discrete fractures was defined across a subregional area measuring about 11 km × 18 km down to a depth of 1 500 m. The fracture network is a statistical representation of fractures based on surface lineament studies for a Canadian Shield area, and consists of a large number of intersecting features within the first few hundred metres with fewer, larger and more vertical features extending to greater depths. Smaller fractures are accounted for in the effective permeabilities assumed for the host rock. The repository is placed at a location where it fits between the assumed fractures at a defined depth of 500 m. A well is placed at the fracture location with the shortest contaminant transport time from repository to surface.

Figure 6 shows the well location and near surface fracture system in the vicinity of the repository.

The repository is shown for context, with surface water features shown in blue. The repository is at a much lower depth where there are no intersecting fractures.

Figure 6: Well location and near surface fracture system



Rock mass hydraulic conductivities are assumed to decrease with depth as shown in Table 1. Fractures are assumed to be uniformly and highly permeable, with a hydraulic conductivity of 4.0×10^{-7} m/s from repository depth to surface.

Table 1: Geosphere hydraulic conductivity

Depth (mBGS)	Thickness (m)	Hydraulic conductivity (m/s)*			
		Reference case	Sensitivity 1	Sensitivity 2	Sensitivity 3
Ground surface to 10 mBGS	10	1×10^{-8}	1×10^{-8}	1×10^{-8}	1×10^{-8}
10 mBGS to 150 mBGS	~140	2×10^{-9}	2×10^{-8}	2×10^{-10}	2×10^{-11}
150 mBGS to 700 mBGS	550	4×10^{-11}	4×10^{-10}	4×10^{-12}	4×10^{-13}
700 mBGS to 1 500 mBGS	800	1×10^{-11}	1×10^{-10}	1×10^{-12}	1×10^{-13}

* Sensitivities 1, 2 and 3 are cases that examine the effect of geosphere permeability.

Scope of the illustrative post-closure safety assessment

Consistent with the specification of CNSC Guide G-320, the scope of work considers both normal evolution and disruptive scenarios. The normal evolution scenario represents the normal (or expected) evolution of the site and facility, while disruptive scenarios examine the effects of unlikely events that might lead to penetration of barriers and abnormal degradation and loss of containment.

Results are measured against criteria for the radiological protection of persons, criteria for the protection of persons from hazardous substances, criteria for the radiological protection of the environment and criteria for the protection of the environment from hazardous substances.

Scenario identification and description

Post-closure safety is assessed through consideration of potential future scenarios, where a scenario is a postulated set of conditions or events. The purpose of scenario identification is to develop a comprehensive range of possible future evolutions against which the performance of the system can be assessed.

Scenarios of interest are identified through consideration of the various features, events and processes (FEP) that could affect the repository system and its evolution. FEP are categorised as either "external" or "internal", depending on whether they are outside or inside the spatial and temporal boundaries of the repository system. Repository and contaminant factors can be considered "internal" factors, whereas the "external" factors originate outside these boundaries. Hence, the repository and contaminant factors are referred to as internal FEP and the external factors are referred to as external FEP.

The external FEP provide the system with boundary conditions and include influences originating outside the repository system that might cause change. Included in this group are decisions related to repository design, operation and closure since these are outside the temporal boundary of the post-closure behaviour of the repository system. If these external FEP can significantly affect the evolution of the system and/or its safety functions of containment and isolation, they are considered scenario-generating FEP (IAEA, 2004) in the sense that whether or not they occur (or the extent to which they occur) could define a particular future scenario that should be considered.

The following sections describe the scenarios that were developed following the FEP review.

Normal evolution scenario

The post-closure safety assessment adopts scientifically informed, physically realistic assumptions for processes and data that are understood and can be justified on the basis of the results of research and/or future site investigation. Where there are high levels of uncertainty associated with processes and data, conservative assumptions are adopted and documented to allow the impacts of uncertainties to be bounded.

The normal evolution scenario is based on a reasoned extrapolation of the reference site and repository characteristics over time. The case study report (NWMO, 2012) describes why the used fuel copper containers are expected to remain intact over the time frame of interest. For the normal evolution scenario, a small number of containers are assumed to be placed in the repository with undetected defects in the copper shell. Conservatively, these containers are assumed to be positioned within a placement room associated with the shortest travel time through the geosphere to the surface biosphere.

Significant features of the reference case of the normal evolution scenario are:

- Waste form characteristics and conceptual repository design as outlined earlier.
- Hypothetical geosphere as outlined earlier.
- Three containers with undetected defects[2] (radius = 1 mm) placed in the repository at the position with the shortest groundwater transit time to the surface.
- No evolution of the defect with time.
- No other container failures occur.

2. Based on a simple estimate of the likelihood of undetected defects arising in the copper shell welding and inspection process (i.e. 1/5000, Maak, *et al.*, 2001).

- Groundwater fills the defective containers 100 years after the containers are placed in the repository.
- Constant temperate climate.
- Self-sufficient farming family growing crops and raising livestock on the surface above the repository.
- Drinking and irrigation water for the family obtained from a 100 m deep well located along the main pathway for contaminants released from the defective containers.

The anticipated effects of glaciations are described by referencing detailed glaciation studies performed for a similar repository with a similar geosphere (Garisto, *et al.*, 2010).

Results are generated for two complementary indicators of radiological safety.

The consequences of gas generation caused by decomposition of organics and corrosion of steel in the defective containers and rock bolts are also determined.

Sensitivity studies

Recognising that there are uncertainties associated with the future evolution of a repository, the assessment has varied a number of important parameters and assumptions, completed bounding assessments and developed a number of hypothetical "What if?" scenarios to explore the influence of parameter and scenario uncertainty in assessing long-term safety. This approach is consistent with CNSC Guide G-320 on the use of different assessment strategies.

Key parameters that could potentially affect long-term safety are varied in deterministic sensitivity cases to understand the impact of uncertainties in these parameters. Some parameters are also pushed beyond the reasonable range of variations in bounding assessments. In these extreme cases, parameters are completely ignored by setting their values to zero or by removing physical limits.

The following lists the deterministic sensitivity cases examined:

- Fuel dissolution rate increased by a factor of 10.
- Instant release fractions set to 0.1 for all radionuclides.
- Container defect area increased by a factor of 10.
- Fracture distance with respect to the defective containers increased and decreased.
- Geosphere hydraulic conductivity changed (increased and decreased from the reference case values).
- Hydraulic conductivity of the excavation damage zone (EDZ) increased by a factor of 10.
- Low sorption in the geosphere with coincident high solubility limits in the container.

The following lists the extreme cases in which parameters were pushed beyond the reasonable range of variation:

- no sorption in the geosphere;
- no solubility limits in the container;
- no sorption in the near field.

Probabilistic analyses

An increased understanding of uncertainties can be obtained through probabilistic modelling. A probabilistic analysis was conducted on the contaminant release and transport parameters, with fixed fracture network and groundwater flows. The parameters varied include diffusion coefficients, sorption coefficients, the number and location of defective containers, the defect size, the size of the critical group and most of the biosphere parameters. A total of 120 000 simulations were examined to identify a 95th percentile peak dose rate.

Disruptive scenarios

The FEP review resulted in identification of the following disruptive scenarios:

- Inadvertent human intrusion.
- All containers fail; a base case with failure at 60 000 years and a sensitivity case with failure occurring at 10 000 years.
- Fracture seal failure and its variant case in which both the fracture seals and the repository tunnel and room seals degrade rapidly and extensively.
- Shaft seal failure.
- Poorly sealed borehole.
- Undetected fault.
- Container failure.

Not all scenarios were analysed in the illustrative case study. The key scenarios considered were “inadvertent human intrusion”, “all containers fail” and “all containers fail with varying rock hydraulic conductivity”. Additional disruptive scenarios would be assessed as part of a safety case for a candidate site.

Methodology overview

The general approach for conducting the post-closure safety assessment is as follows:

1) *Perform radionuclide screening*

   Used fuel contains many hundreds of radionuclides. Screening is performed to identify the potentially radiologically significant radionuclides so that subsequent work need only consider this group.

   In this study, 39 radionuclides have been considered for the radiological hazard assessment, with these being a mix of long-lived fission products and actinides. A further 37 elements and their precursors have been considered in the non-radiological hazard assessment.

2) *Perform detailed 3-D groundwater flow and radionuclide transport modelling*

   Detailed 3-D steady-state hydrogeological modelling is performed with the FRAC3DVS-OPG code to determine the groundwater flow field near the repository. FRAC3DVS-OPG is the reference groundwater flow and groundwater transport code used by the NWMO.

   FRAC3DVS-OPG is described in Therrien, *et al.* (2010).

   Once the flow field is determined, detailed 3-D radionuclide transport calculations (accounting for diffusive and advective processes) are performed for a small group of radionuclides (i.e. ^{129}I, ^{36}Cl, ^{41}Ca, ^{234}U and ^{238}U) that represent a range of low-sorption to high-sorption species. These calculations aid in understanding the transport behaviour and provide data for subsequent use in SYVAC3-CC4 system modelling.

The FRAC3DVS-OPG code does not have a biosphere model so another code must be used to determine the dose consequences.

3) *Perform system modelling*

SYVAC3-CC4 is a simple model used for radionuclide transport and the calculation of dose consequences. The code simulates the used fuel, container, vault, geosphere and biosphere, allowing feedback between these components, e.g. the geosphere groundwater flows can be affected by the well demand established by the biosphere model. SYVAC3-CC4 is very fast (in comparison to FRAC3DVS-OPG) and is used to perform both deterministic and probabilistic calculations.

SYVAC3-CC4 is described in NWMO (2011).

The 3-D groundwater advective flow field generated with FRAC3DVS-OPG is used to develop a network of 1-D geosphere transport conduits. To provide confidence in the resulting model, radionuclide transport calculations are performed for ^{129}I, ^{36}Cl, ^{41}Ca, ^{234}U and ^{238}U and compared to similar results obtained from the more detailed FRAC3DVS-OPG model.

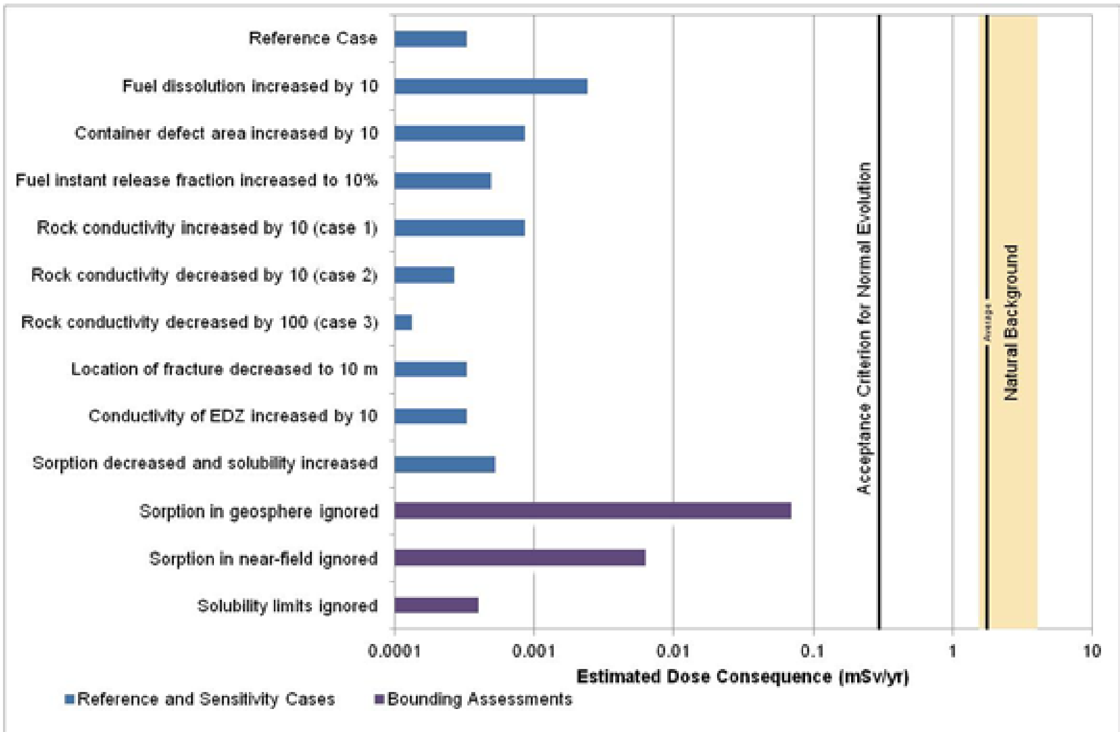
The code is used to determine the consequences for the reference case and the various deterministic and probabilistic sensitivity cases. In the dose assessment simulations, the full suite of 39 radionuclides of potential interest is considered.

Results

Normal evolution scenario

Figure 7 shows the results obtained for the reference case normal evolution scenario along with its associated sensitivity cases.

Figure 7: Normal evolution cases



The primary contributor to the public dose over the long term from the three assumed defective used fuel containers is ^{129}I, a long-lived fission product that is non-sorbing in the geosphere. The calculated peak dose for the reference case is about 900 times lower than the interim dose acceptance criterion of 0.3 mSv per year and occurs at about 100 000 years after closure. The long time to peak dose is due, in part, to the combined performance of the repository barrier systems which includes the robustness of the long-lived containers, the integrity of the engineered sealing systems and the near-field rock surrounding the repository.

The sensitivity analyses show that the impact on dose is small when key parameters are varied over credible ranges. The parameter with the most significant impact on dose is the dissolution rate of the used fuel. This is because the bulk of the radionuclides are retained within the fuel matrix over the time frame of interest. The bounding assessments show a noticeable increase in dose when sorption is completely ignored. A small amount of sorption can make a significant difference to some radionuclides, particularly ^{238}U and its daughters (i.e. ^{210}Po, ^{222}Rn, ^{210}Pb and ^{226}Ra).

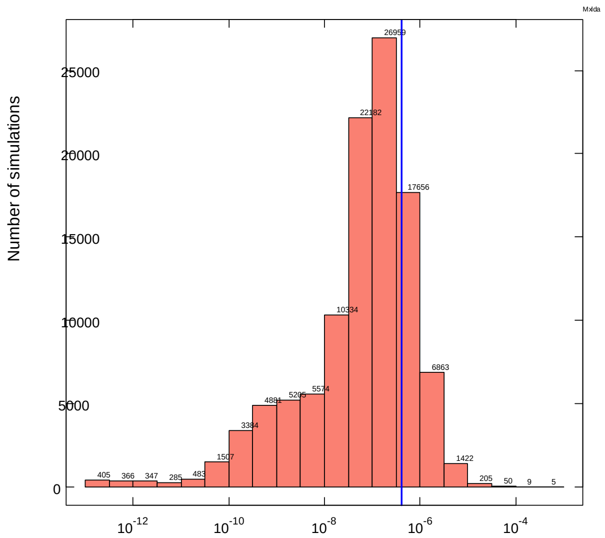
Recognising the importance of parameters such as used fuel dissolution and sorption as shown in this case study, NWMO maintains active research programmes in these areas to continue to improve understanding of these processes (Shoesmith, 2007; Vilks, 2011).

For non-radiological hazards, the results indicate that the large amounts of uranium and copper in the repository do not pose a risk. Hg, Cd and Pb had the highest concentrations relative to criteria, but remained well below the interim acceptance levels.

Figure 8 shows the results of the probabilistic assessment. The 95th percentile dose consequence in this case is assessed to be 4 times greater than the reference case, but still much less than the dose criterion.

Figure 8: Probabilistic results

The blue line is the average dose



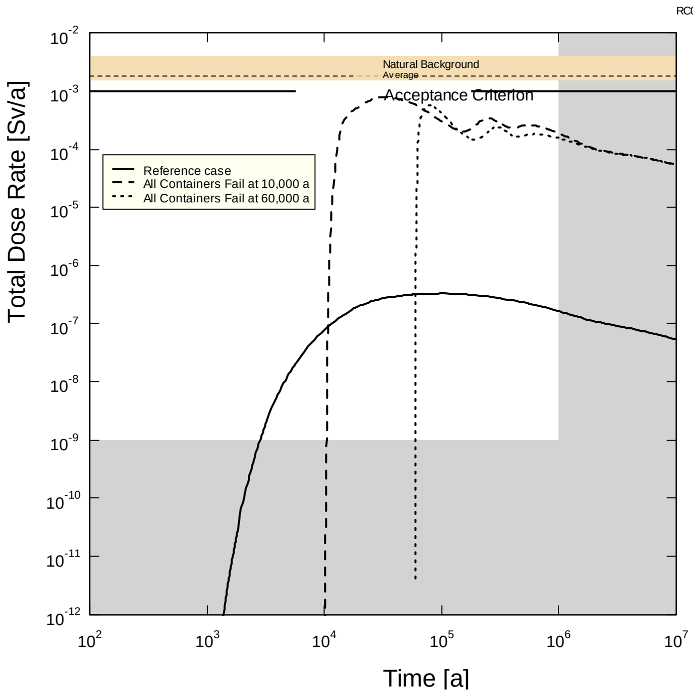
Disruptive scenarios

Figure 9 shows the results for the “all containers fail” scenarios.

The results indicate a significant increase in the peak dose results; however, the peak doses remain below the interim dose acceptance criterion of 1 mSv per year for disruptive scenarios. The results also indicate that there is low sensitivity of peak dose to the assumed failure time in the reference geosphere. This occurs primarily as the assumed container failure time is longer than the short-lived fission product decay time. The remaining actinides and most of the remaining long-lived fission products are sorbed within the clay-based engineered barriers and natural barrier of the enclosing rock mass so that the peak dose rate does not substantially change between these two cases.

The peak results are found to be more sensitive when varying the geosphere hydraulic conductivity at the same time as having all the containers fail at 10 000 years. When the hydraulic conductivity is increased by a factor of 10, the peak dose rate occurs at 17 000 years and is at least a factor of 1.7 times above the disruptive events interim dose rate acceptance criterion. This result suggests that a candidate site with a rock mass hydraulic conductivity on the order of 4 ´ 10^{-10} m/s or higher would have the potential to exceed the interim dose acceptance criterion for disruptive scenarios.

Figure 9: Results for “all containers fail”



The “inadvertent human intrusion” scenario is a special case, as recognised in CNSC Guide G-320, since it bypasses all the multiple barriers put in place, and therefore the associated dose consequence could exceed the regulatory dose limit. The results show a potential dose from early intrusion to the drill crew, and to a conservatively defined site resident, exceeds the dose limit. However, the likelihood of this event occurring is very small due to placing the used fuel containers deep underground, with institutional controls, no economically viable mineral resources, and no potable groundwater resources. Normal deep drilling practices (e.g. control of drilling fluids, use of gamma logging, etc.) will also tend to reduce consequences relative to those estimated here. Although the likelihood of human intrusion cannot be readily defined, it will be very low. The annual risk of health effects from human intrusion is estimated to be less than 1 in 100 000 per year.

Conclusions

The illustrative case study describes the reference design for a deep geological repository in crystalline rock and provides an illustrative post-closure safety assessment approach, which is structured, systematic and consistent with CNSC Guide G-320. The assessment includes a description of the repository system, systematically identifies scenarios, models and methods for evaluating safety, uses different assessment strategies, addresses uncertainty, and compares the results of the assessment with interim acceptance criteria.

The post-closure safety assessment shows, for the normal evolution scenario and associated sensitivity cases, that all radiological and non-radiological interim acceptance criteria are met with substantial margins during the post-closure period. This result is consistent with previous assessments of a deep geological repository in Canada, as well as with safety assessment studies by other national radioactive waste management organisations.

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Performance assessment and the safety case: Lessons from recent international projects and areas for further development

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Abstract

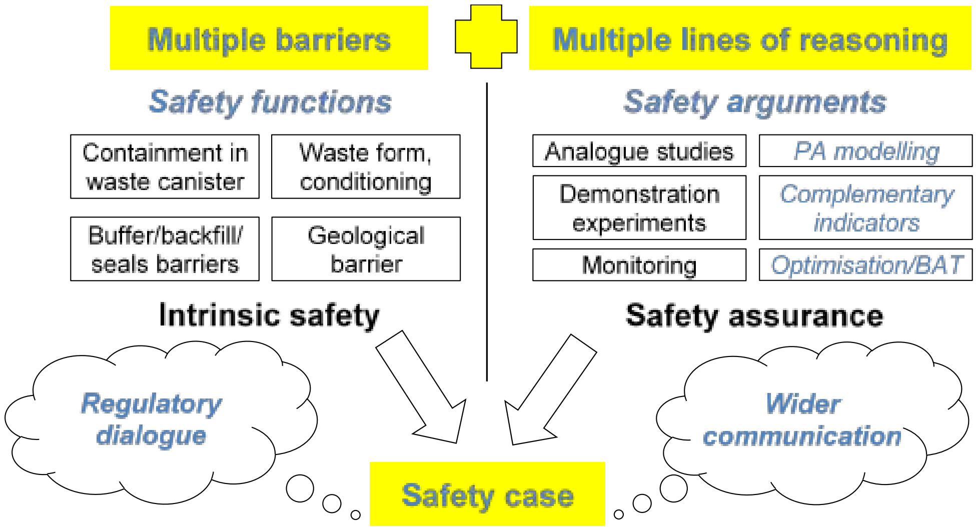
The European Commission (EC) PAMINA project – Performance Assessment Methodologies in Application to Guide the Development of the Safety Case – was conducted over the period 2006-2009 and brought together 27 organisations from 10 countries. PAMINA had the aim of improving and developing a common understanding of performance assessment (PA) methodologies for disposal concepts for spent fuel and other long-lived radioactive wastes in a range of geological environments. This was followed by a Nuclear Energy Agency (NEA) sponsored project on Methods for Safety Assessment of Geological Disposal Facilities for Radioactive Waste (MeSA), which was completed in 2012. This paper presents a selection of conclusions from these projects, in the context of general understanding developed on what would constitute an acceptable safety case for a geological disposal facility, and outlines areas for further development. The paper also introduces a new project on PA that is under consideration within the context of the EC Implementing Geological Disposal of Radioactive Waste Technology Platform (IGD-TP).

Introduction to the safety case

A safety case for a geological disposal facility (GDF) is a set of claims concerning the safety of the disposal of radioactive waste, substantiated by a structured collection of arguments and evidence. Within the safety case, the performance of the facility against quantitative safety standards is evaluated through calculations. A quantitative safety assessment involves developing an understanding of: *i)* how safety functions contribute to isolation and containment of the waste; *ii)* how, and under what circumstances, contaminants might be released from a GDF; *iii)* how likely such releases are; *iv)* what the potential radiological or other consequences of such releases could be to humans and the environment. Importantly, it is necessary to understand how the geological characteristics of the site and the components of the design will evolve and function, and document the uncertainties associated with the assessment and their potential consequences. The term "PA" is used more generally to refer to analyses of the performance of the overall geological disposal system or of particular sub-systems.

Figure 1: Some key features of a safety case

Items in italics were addressed within PAMINA and MeSA



Some key features of a safety case are illustrated in Figure 1. A safety case needs to bring together and effectively integrate a wide range of safety arguments and analyses. Safety cases require siting and design strategy, an assessment strategy and a management strategy. The siting and design strategy describes how the system will provide safety through the use of multiple engineered and natural barriers. The assessment strategy describes how the requisite level of safety can be demonstrated, and needs to make use of both quantitative and qualitative lines of reasoning. Neither strategy should rely unduly on any single component – whether that is a physical component of the system (such as a particular engineered barrier) or a single element of the assessment strategy (such as numerical modelling). The management strategy should ensure that siting and design and assessment strategies are implemented with the appropriate degree of co-ordination and quality.

A safety case must be underpinned by the results from research and development (R&D) studies, design and site characterisation, and demonstration of how regulatory requirements and guidance have been or will be met. The safety case will need to be developed in a staged manner, consistent with a staged approach to GDF conceptual development, feasibility studies, site selection and characterisation, licensing, construction, operational testing, full-scale operation and closure. A safety case needs to be informed at each development stage by dialogue with regulators and stakeholders.

PAMINA and MeSA projects

The parts of the safety case considered within the EC PAMINA project are illustrated by the use italics in Figure 1 (safety functions; safety arguments and the use of PA modelling, complementary performance indicators, and optimisation/best available techniques (BAT); regulatory dialogue; wider communication). The work within PAMINA was organised into four research and technology development components (RTDC) having the following main aims:

- *RTDC-1:* To evaluate the state of the art by undertaking a comprehensive review of PA methodologies and experience in participating organisations.
- *RTDC-2:* To establish a framework and methodology for the treatment of uncertainty in PA and safety case development, and to document good practice.

- *RTDC-3:* To develop and improve particular PA methods and tools.
- *RTDC-4:* To conduct collaborative PA exercises designed to understand the potential implications of undertaking modelling at different levels of process and geometric complexity.

The results of RTDC-1 constitute the *European Handbook of Safety Assessment Methods for Geological Repositories – Part 1*, while the results of RTDC-2, RTDC-3 and RTDC-4 collectively form the *European Handbook of Safety Assessment Methods for Geological Repositories – Part 2*. The European Handbook is therefore the key output of the project. The main introduction to the project and its results is via the *PAMINA Project Summary Report* (Galson, 2011).

The *European Handbook – Part 1* (Bailey, 2011) is based on reviews conducted of the state of the art as of the start of the project, and is divided into 11 topic areas: *i)* safety functions; *ii)* definition and assessment of scenarios; *iii)* safety indicators and performance/ function indicators; *iv)* uncertainty management and uncertainty analysis; *v)* safety strategy; *vi)* analysis of system evolution; *vii)* sensitivity analysis; *viii)* modelling strategy; *ix)* human intrusion; *x)* biosphere modelling; *xi)* criteria for data selection/input. The other three RTDCs took forward work in all of these topic areas.

Subsequent to PAMINA, further dialogue on some of these issues was undertaken at the international level within the context of the NEA MeSA project, which enabled the thinking within PAMINA to be further shared and integrated, and a wider consensus documented (OECD/NEA, 2012). Seven assessment strategy issues were considered with the MeSA project: *i)* safety assessment in the context of the safety case; *ii)* safety assessment and safety case flowcharts; *iii)* system description and scenarios; *iv)* modelling strategy; *v)* safety assessment indicators; *vi)* treatment of uncertainties; *vii)* regulatory issues.

Selected conclusions from PAMINA and MeSA

Key conclusions from PAMINA and MeSA include:

- Safety assessment forms a central part of the safety case. However, the results of such assessments must be placed in context and augmented by additional information in a safety case to support decision making. Whereas in the past, safety case development placed a lot of emphasis on comparison between safety assessment calculation results and dose/risk criteria set by the regulator, recent safety cases have used a broader range of performance indicators and safety arguments. BAT, optimisation, safety functions and alternative safety and performance indicators are increasingly being used as additional arguments in a safety case in support of compliance with the regulatory dose/risk criteria and to build confidence in the long-term safety and the robustness of GDF design options. Doses and risks remain the primary safety indicators, but it is understood that over long time scales such calculations should be considered as illustrative.
- The concept of using various types of indicators to complement dose and risk has developed considerably during the last 20 years, although terminology is not consistent between national programmes. Calculation of a range of alternative safety and performance indicators beyond the traditional dose/risk approach can assist in demonstrating safety, understanding of subsystem performance, and building confidence in the multi-barrier approach and optimisation decisions. It can also assist in wider communication of the safety case when addressing both technical and lay audiences. This does not remove the need to provide detailed calculations to regulatory authorities for comparison to regulatory dose/risk performance measures, but alternative indicators provide a useful adjunct.

- The main focus of safety assessment remains an evaluation of radiological impacts on humans, but there is an increasing recognition of a need for consideration of the potential impacts on non-human biota. In addition, the potential impacts of chemotoxic elements in the wastes are regulated, e.g. by groundwater protection legislation, and there is increasing work to develop approaches to demonstrate compliance with such legislation.
- Safety assessment provides information to focus research and development, site characterisation and design. Conversely, these aspects of GDF development produce the data and interpretations that support assessments. It is therefore essential to ensure clear and effective information flow between the parties involved with GDF development. As programmes mature, safety case development is being driven increasingly by two elements. One element is the requirement for staged updating at key programme decision stages – where decisions are likely to be based on a much wider range of factors than purely safety arguments, such as the need to demonstrate optimisation and the use of BAT. Second, the development and implementation of an assessment approach based on the use of safety functions that tie into the multi-barrier approach is being increasingly used as a means to structure assessments and to communicate the outputs.
- Uncertainties are, and always will be, associated with assessment results. Internationally, there is a high level of consensus on the types and sources of uncertainties in safety assessments. Typically the uncertainties considered in safety assessment are classified into scenario uncertainties, model uncertainties, and data or parameter uncertainties. Strategies for treating uncertainties within the safety assessment are well established.
- Scenarios represent specific descriptions of a potential evolution of the disposal system from a given initial state. They describe the compilation and coupling of safety-relevant features, events and processes (FEP) as a fundamental basis for the assessment of post-closure safety. The development of scenarios constitutes a key element of the management of uncertainties. Catalogues of FEP describing all of the possible influences within and on the disposal system are seen as useful in driving or auditing the development of expected evolution (or base case) scenarios and altered evolution (or variant) scenarios for use in PA. Scenarios are increasingly being developed by consideration of how particular FEP could affect the safety functions of a disposal system. However, the FEP to be considered were largely identified through structured elicitation exercises conducted in the 1990s, and there has been little effort to develop fundamentally new FEP databases since then.
- The main consideration in the assignment of probabilities to scenario-forming FEP is safety case robustness and credibility. Where statistical information is available, this should be used. Otherwise, probabilities should be assigned on a cautious basis and should be avoided where regulatory guidance provides for this; where insufficient information is available; where assessment outcomes do not depend on this probability; or where siting has already explicitly considered the uncertainty and there is little that can be done to reduce the probability further (e.g. future human intrusion, see below). Where formal expert elicitation is used to define probabilities, it is important to record the experts' thinking and to identify any factors that could affect probability estimates, in order to demonstrate transparency in attributing probabilities to particular parameters or events. Use of formal methods may be justified where safety case outcomes rely significantly on probability estimates. Robustness and credibility can be enhanced by careful explanation that most so-called "probabilities" are actually "degrees of belief" or "weightings", rather than formal mathematical probabilities. Such a treatment means that it is permissible to assign scenario weightings that total more than one. This allows for a robust treatment of scenario uncertainty in a PA, even if not consistent with a purely mathematical treatment of scenario probabilities.

- As there is little scientific basis for predicting the nature or probability of human actions in the far future, the safety case for a GDF should focus on the potential consequences of inadvertent intrusion using one or more stylised scenarios. In contrast to the assessment of naturally occurring FEP, such analyses need not aim for comprehensiveness. The range of possible future human actions is large, and it is more appropriate to evaluate the resilience of disposal system design to stylised events. Some national regulations include requirements on how inadvertent human intrusion should be treated in assessments.
- Overall, there is wide consensus on modelling strategies to support safety assessment. There is significant interest in developing more complex models to represent the different components of the disposal system as programmes mature, in order to demonstrate adequate knowledge and capability to evaluate system behaviour over time and to assist with design optimisation. Comparisons between models having greater and lesser geometric and process complexity have demonstrated that in the early stages of a GDF development process, simplified models can be successfully used to provide an indication of where more detailed investigations are required. As the programme matures, more complex models are likely to become available. If the results obtained using a complex model with many parameters can be reproduced using a simple model with a few parameters, it is clear that the key processes and parameters (those included in the simplified model) have been identified and the system is reasonably well understood. This would be a strong argument in a safety case.
- Whether conservative or best-estimate assumptions and parameter values are used in a PA, and whether deterministic and/or probabilistic calculation methods are used, these should be based on a transparent use of expert judgement. When combined with a clear audit trail, this will allow regulators and other interested stakeholders to better understand the potential impact on safety posed by model, parameter and/or scenario uncertainties, and the way in which these have been addressed. Guidance has been developed within PAMINA on good practice for formal expert elicitation and the treatment of parameter and model uncertainties, the use of which can introduce a higher level of consistency and confidence in assessment outcomes and the safety case.
- Sensitivity analysis is an important tool in understanding the impacts of particular model inputs on the overall safety of the disposal system, and allows effort and investigations to focus on those parameters, models and scenarios that have the greatest potential impacts on safety. Comparisons of sensitivity analysis approaches using both synthetic problems and real data from ongoing site-specific investigations have shown that the current level of capability amongst those working in the field is high, and adds to the confidence that suitable models and analytical approaches are available. Guidance has been provided within PAMINA on what techniques are most suitable for use in particular circumstances.
- Spatial variability of parameter values can have considerable impact on the understanding of subsystem performance and the safety functions ascribed to sub-systems, such as mechanical stability and the ability of the geosphere to retard migrating radionuclides. There is a need for further work concerning the difficulties of transforming individual measurements of safety-related parameters, such as fluid flow rates and hydraulic conductivity, into parameter values that can be used with greater justification in large-scale radionuclide migration models. Examination within PAMINA of a new approach to simulate radionuclide transport as a sequence of particle transfer rates (Continuous Time Random Walk) indicates that this could offer an effective means to quantify radionuclide transport in a wide range of porous and fractured media.

- The maturity and complexity of biosphere modelling approaches and dose assessment strategies differs between organisations in different countries, mainly due to differences in national regulatory frameworks and differences in the maturity/timing of the programmes. Issues associated with biosphere modelling required for long-term assessments of radioactive waste disposal have been dealt with in greater detail in other international projects (e.g. BIOPROTA – www.bioprota.org).
- PAMINA included a workshop on the regulatory perspective to PA and the safety case. Some of the high-level conclusions are that dose-based regulatory criteria should avoid language that discourages a developer/operator from exploring the full range of uncertainty owing to a concern that some calculations might yield results exceeding the criteria. Risk-based criteria should not be limited to requesting the presentation of mean values, but should encourage the developer/ operator to discuss and present the entire range of uncertainty. Given that long-term calculated doses are interpreted more as illustrative performance measures, the validity of basing regulatory decisions on the use of a dose "limit" for the long term is questionable. This line of thinking has led to significant regulatory redevelopment in a number of countries over the last 10-15 years. Second-generation regulations for GDF explicitly recognise the implications of the long assessment time frame for the demonstration of compliance, and take explicit account of the wider understanding developed within the "safety case" and the importance of concepts such as optimisation, BAT and safety functions in driving decision making. Regulators expect not only an assessment of compliance with quantitative radiological criteria, but also a demonstration that the geological disposal system is robust and that its possible evolution is well understood. Quality assurance, quality management and transparency and traceability of the assessment process are essential to building an acceptable safety case. However, given the long time scales of GDF development programmes, regulators will continue to learn, and future regulatory guidance will increasingly be informed by the national GDF development programmes.

Where are we now and where next?

There is an increasing databank of national safety assessments, built up over the last 25 years or so. Already some 20 years ago the NEA Radioactive Waste Management Committee, jointly with the International Atomic Energy Agency and the EC, were collectively able to:

Confirm that safety assessment methods are available today to evaluate adequately the potential long-term radiological impacts of a carefully designed radioactive waste disposal system on humans and the environment, and

Consider that appropriate use of safety assessment methods, coupled with sufficient information from proposed disposal sites, can provide the technical basis to decide whether specific disposal systems would offer to society a satisfactory level of safety for both current and future generations. (OECD/NEA, 1991)

There is good knowledge of the sources of uncertainty in PA and the safety case, and how they can be managed through a multi-factor safety case and multiple lines of both qualitative and quantitative reasoning – as evidenced in *PAMINA RTDC-2* (Crawford, 2009). There is also a good understanding that expert judgement runs through all steps of a PA and the safety case, and that systematic approaches are needed to develop the PA and the wider safety case – as evidenced in the *European Handbook – Part 1* (Bailey, 2011). Finally, there is good understanding of the issues involved in regulating a project that requires safety to be considered over many thousands of years.

PA and safety case topics that would benefit from further development within national programmes include the following:

- The management of PA and integration of PA activities with other parts of the disposal programme as the programme matures. In particular, there is a need to better understand: *i)* how PA methods can be used to support optimisation of design as a programme moves closer to actual implementation; *ii)* how PA can be used as a tool to help inform and prioritise investigation and R&D studies; *iii)* the appropriate balance between quantitative PA methods and "complementary considerations" in a safety case; *iv)* how to maintain traceability of how and why a PA and the safety case evolve as disposal programmes move forward over many decades.

- The communication of PA and the safety case to stakeholders having different degrees of understanding and/or different frameworks for understanding about long-term safety issues.

- Further refinements to PA tools – for example increasing use of fast-running system assessment models, supported by more detailed component models, to assist decision making with such issues as optimisation, disposal layout, waste packaging proposals and development of waste acceptance criteria. There has also been renewed interest internationally in the use and development of total system probabilistic assessment approaches, as an adjunct to mainly deterministic approaches, to better understand model sensitivities.

In addition, we note that even as more data and understanding accrue, there will always be uncertainties that remain to be managed in the safety case, particularly those that are present over long time scales. There will be a need to demonstrate that such residual uncertainties are unimportant. However, the regulator will always have the job of making a decision in the face of these uncertainties. Further regulatory development and guidance can be expected on PA and the safety case as programmes mature and as regulators learn along with the teams developing the PA/safety case.

The MeSA project led to several suggestions on areas related to the safety case in which further development work might be conducted by the NEA. These included suggestions: *i)* to update the NEA brochure on the safety case concept; *ii)* to update the NEA FEP database; *iii)* to initiate a project on exchange of information and best practice on scenario development; *iv)* to develop a state-of-the-art report on safety indicators; *v)* to develop guidance on conduct of sensitivity analyses; *vi)* to develop guidance on when formal approaches to expert judgement and elicitation may be warranted in safety assessment.

Another international initiative is now being pursued in the context of the IGD-TP. One of the strategic research topics within the IGD-TP concerns the safety case. Work in three topic areas is foreseen (IGD-TP, 2012):

1) Increasing confidence in the research models used to underpin safety assessments. The objective is to test and compare coupled material interaction models.

2) Improving communication of the safety case. The objectives are to create and maintain a resource pool of experts available for review of scientific and technical reports, and to create a channel for dialogue on research into the post-closure safety of geological disposal.

3) Increasing confidence in methods for the management of uncertainty in safety assessments. The objective is to evaluate the state of the art of the different methods used, and to refine them for practical application in safety assessments in support of licensing.

A technical/scientific working group has been established, and international meetings were held in May and September 2013 to consider a possible new collaborative project in the third area, building on the achievements of PAMINA and MeSA. Topics under consideration after the September meeting include:

- *the management of uncertainties*, including issues such as overall strategies, management of uncertainties in different time frames of disposal system evolution, regulatory decision-making under uncertainty and communication of uncertainty to different stakeholders;
- *uncertainty identification and quantification*, including issues such as expert judgement, derivation of PDF, and identification and quantification of correlated parameters;
- *sensitivity analysis*, including such issues as survey and evaluation of methods suitable for use in assessments, comparison of methods in numerical experiments and use of sensitivity analysis to guide R&D programmes.

It is likely to be a couple of years before this becomes an EC project, but there is planned to be some collaboration on specific technical topics before then.

Acknowledgements

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Session 7.2

Scientific and Technical Basis

Applicability of indicators in clay and salt representing the safety function containment

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The last several years have seen a number of important developments in the use of indicators in safety cases for geological disposal, e.g. Becker, *et al.* (2009). The NEA has reviewed these developments within the scope of its Methods for Safety Assessment (MeSA) project. The review confirmed that indicators complementary to dose and risk are now accepted by the majority of regulators and implementers as an important component of a safety case. It is likely that all future safety cases will integrate one or more complementary indicators in their evaluation of repository safety and performance (OECD/NEA, 2012). Two important conclusions can be drawn from the review:

- One of the primary drivers behind the use of complementary indicators is that they can offset some of the drawbacks associated with the calculation of dose and risk. In particular, assumptions to be made about future biosphere processes can be avoided and they allow the performance of the repository to be disaggregated rather than presented as a single parameter thus enabling the performance and safety functions of individual barriers or repository components to be assessed.

- A further trend is to apply indicators that are derived from safety functions. Safety functions play an important role in constituting the safety concept and safety principles. Indicators that can quantify the role of the safety functions are important arguments for the whole safety case.

Applied indicators

An important safety function of repositories for the final disposal of radioactive waste is to isolate and contain the waste to ensure long-term safety of people and protection of the environment. Isolation is provided by the repository's depth. There are many different designs for repositories but containment is generally provided by geological and a series of engineered barriers. For example, in the German safety requirements (BMU, 2010) containment "refers to a safety function of the repository system which is characterised by the fact that the radioactive waste is contained inside a defined rock zone, in such a way that it essentially remains at the site of emplacement, and at best, minimal defined quantities of material may leave that rock zone." An important element of this approach is the definition of the rock zone, which in conjunction with the technical seals, e.g. shaft seals or backfill, ensure containment of the waste. This rock zone is designated as a "containment-providing rock zone" (CRZ).

One approach to define a set of indicators that supports the use of the CRZ and thus is directed to the safety function containment is provided by Baltes, *et al.* (2007). The proposed set of indicators comprises five indicators complementary to the effective dose:

1) *Proportion of the cumulative released quantity of substance over the safety case period.* The first proposed indicator is the ratio of the total released radionuclide quantity from the CRZ to the initially emplaced total amount of radionuclides. This definition implies that this indicator cannot have a direct safety statement, since it does not assess the radiological consequences of the released radionuclides. Still, this indicator can be used to illustrate what quantities of radionuclides are released from the CRZ and how small the percentage of released radionuclides is in comparison with the emplaced inventory. The yardstick proposed by Baltes, *et al.* (2007) requires that less than 0.01 mol-% of the total radionuclide inventory will be released within the assessment period of 10^6 years.

2) *Concentrations of released U and Th in the porewater at the CRZ boundary.* The idea of this indicator is to illustrate how small the percentage of the radionuclide concentration caused by release from the repository is in comparison with naturally occurring radionuclide concentrations. Therefore, the concentration of all released radionuclides of uranium and thorium is used for the comparison to the natural concentration of these elements. The proposed yardsticks are 1 g/l for uranium, and 0.1 g/l for thorium (Baltes, *et al.*, 2007).

3) *Contribution to power density in the porewater at the CRZ boundary.* Starting point for the calculation of this safety indicator is the calculated activity concentration in the groundwater at the boundary of the CRZ. Actually, it can be the activity concentration in any subsystem of the repository system, but it is proposed in Baltes, *et al.* (2007) to use the power density in the porewater and in the soil matrix in the deeper aquifer system. Here, the indicator is calculated from the activity concentration in the porewater without considering the power density coming from radionuclides in the soil matrix, since no data for the matrix are available. The calculation of the power density is carried out with a simple weighting scheme by multiplying the activity concentration of every radionuclide [Bq/m^3] in the porewater at the boundary of the CRZ with its decay energy. This operation yields a power density p (power per volume, [MeV/(s×m^3)]):

$$p = \sum_{\text{all nuclides } n} c_n E_n$$

with the activity concentration c_n [Bq/m^3] of radionuclide n in groundwater and the corresponding decay energies E_n [MeV].

The indicator contribution to the power density in groundwater is independent of any specific biological species. It is an aggregation over all radionuclides and can be seen as a yardstick for the impact on biota in general. It is difficult to derive such a yardstick for the power density in the porewater and in the soil matrix of a deeper groundwater system. Baltes, *et al.* (2007) estimated a value of 100 MeV/(s×m^3) for the power density in clay. For the upper groundwater above the salt dome Gorleben Becker, *et al.* (2009) derived a natural value of about 75 MeV/(s×m^3).

4) *Contribution to radiotoxicity in groundwater.* The contribution to radiotoxicity in groundwater r is calculated from the radionuclide flux s_n through the boundary of the CRZ. This flux is distributed in a virtual water body used by a group of humans. The virtual water body comprises the annual water consumption (drinking water, irrigation, etc.) of this group. A reference group of 30 humans with an annual water consumption w of 15 000 m^3 is proposed by Baltes, *et al.* (2007) and used for the calculation. To assess the radiological consequences of the radionuclide concentrations in this water body, the same dose conversion factors D_n are applied as for the calculation of the individual dose:

$$r = \sum_{\text{all nuclides } n} \frac{s_n D_n}{w}$$

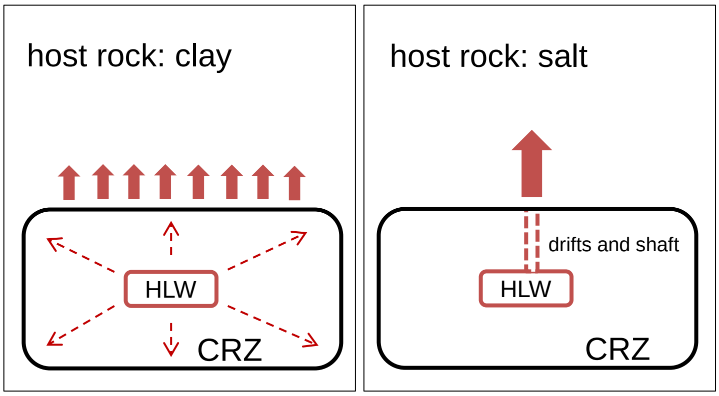
The yardstick should be a small percentage of the average natural background radiation. In Baltes, *et al.* (2007) a yardstick of 0.1 mSv/a is proposed.

5) *Radionuclide concentration in the usable water near the surface.* For radionuclides that are part of natural waters, it is possible to compare the natural concentrations with the concentrations released from the repository into these waters. Therefore Baltes, *et al.* (2007) proposes limits for naturally occurring radionuclides, which can be used for these comparisons. The considered radionuclides and the corresponding yardsticks are as follows:

^{238}U	^{234}U	^{226}Ra	^{210}Pb	^{235}U	^{228}Th	^{230}Th	^{232}Th
1.58 Bq/m^3	1.83 Bq/m^3	0.48 Bq/m^3	0.57 Bq/m^3	0.1 Bq/m^3	0.01 Bq/m^3	0.2 Bq/m^3	0.05 Bq/m^3

These five indicators are proposed, but not tested, in Baltes, *et al.* (2007). Therefore Noseck, *et al.* (2012) evaluated these indicators regarding their usefulness indicating containment for a generic repository system in a clay and a salt formation. The major results are summarised in this contribution. For salt a generic repository system in a salt dome in Northern Germany is used. The applied system is based on the case developed in the context of the research project ISIBEL (Buhmann, *et al.*, 2008). The generic repository system in a clay formation in Northern Germany was derived in the research project TONI (Rübel, *et al.*, 2007). Both projects were carried out independently and differ in some fundamental aspects, e.g. the emplaced inventory (the clay case considers only direct disposal of spent fuel, whereas the salt case considers spent fuel and reprocessed waste). Therefore, the calculations cannot be used to compare both generic sites. The different properties of the repository systems in clay and salt are also important for the evaluation of the release out of the CRZ: In spite of the low permeability of undisturbed clay, radionuclides are generally released by diffusion. This release occurs over the whole area of the CRZ. Permeability of undisturbed salt rock is so low that even diffusion processes will not lead to release out of the CRZ. Therefore in salt only a release via the backfilled drifts and the shaft has to be considered (Figure 1). These significant differences in the release areas between clay and salt have a tremendous effect on the indicators that are used to compare concentrations or dependent quantities calculated directly at the CRZ

Figure 1: Schematic illustration for the release of radionuclides out of the CRZ for a repository in clay and salt



Note: The exact location of the CRZ within the repository system in salt and clay is explained in Noseck, *et al.* (2012).

boundary, namely the concentration of released U and Th and the contribution to the power density in the porewater at the CRZ boundary. As a consequence, the results of these indicators are for rock salt several orders of magnitude higher than in the case of the clay formation, whereas the indicators' radiotoxicity flux and the effective individual dose, which are calculated after dilution in surface near aquifers are within a similar range of magnitude for both formations. Therefore, maximum information for such indicators is received, when they are calculated for both: *i)* referred to the area over which they are released; *ii)* referred to the total area of the rim of the CRZ (Resele, *et al.*, 2010).

Assessment of indicators

The assessment is focused on the function of the indicator and its usefulness indicating containment for two different generic repository systems.

1) For the indicator proportion of the cumulative released quantity of substance over the safety case period the yardstick is fulfilled. The released quantities are only a small fraction of the emplaced waste. The relation of released to total inventory for the considered repository after one million years is 3.85×10^{-12}. As stated above, this does not indicate the potential health hazard of the released radionuclides. Since for this indicator the total amount of all radionuclides in mol is the basis for a repository containing spent fuel, the release of ^{238}U is of crucial importance, since its inventory in SNF is more than two orders of magnitude higher than the amount of each other radionuclide. Therefore, this indicator is not sensitive enough to judge the relevance of the release of nuclides other than ^{238}U.

   This indicator is applicable to both clay and rock salt. It would be a very useful indicator, if it is considered nuclide-specific. Further, a cumulative representation would be interesting, particularly to illustrate the retardation capacity (see Figure 2).

2) The indicator concentration of released uranium and thorium in the porewater at the CRZ boundary considers the radionuclide flow of the natural radionuclides of uranium and thorium, which are usually not relevant. More important are non-natural radionuclides or those occurring in only extremely low concentrations in natural waters such as ^{14}C and ^{36}Cl. Since the radionuclides of uranium and thorium are strongly sorbing, this indicator is not sensitive for repositories in clay formations, where it delivers a value of 0. It is recommended to use the above-mentioned indicator proportion of the cumulative released quantity of substance over the safety case period to illustrate the retardation capacity of the CRZ. According to this the indicator concentration of released uranium and thorium in the porewater at the CRZ boundary does not deliver additional information. It is not recommended for further use.

3) The indicator contribution to power density in porewater at the CRZ boundary is not related to safety, since it is only based on physical quantities. Therefore, it is not dependent on any biosphere or near surface processes and is not covered with the uncertainties existing for these processes. On the other hand it is difficult to derive a yardstick for this indicator. As discussed above for this indicator it is important to define whether an average value normalised to the area of the CRZ boundary or the maximum value referred to the fraction of the surface where the radionuclides are released is applied. We would recommend both applications of the indicator. The maximum value is strongly dependent on the repository formation or concept and allows the identification of different kind of releases, as observed for repositories in clay and rock salt.

4) If an appropriate calculation scheme and an accepted yardstick are applied for the calculation (for example by a given scheme from the regulator) the indicator contribution to radiotoxicity in groundwater gives a strong argument for the safety of the repository system independent of the barrier function of geological formations

outside the CRZ (Figures 3 and 4). Figure 3 illustrates that the contribution to radiotoxicity in salt is dominated by ^{126}Sn with a highest value of about 6.3×10^{-8} Sv/a within 10^6 years, mainly caused by its very high dose conversion coefficient. At later times the radiotoxicity flux is dominated by ^{129}I with the highest value at 8.2×10^6 years of 3.5×10^{-6} Sv/a. Additional important contributors are long-lived nuclides with high dose conversion factors (^{1335}Cs, ^{229}Th). In clay the contribution to radiotoxicity is dominated by ^{79}Se with a radiotoxicity flux of about 10^{-7} Sv/a. At later times the radiotoxicity flux is dominated by ^{129}I. The highest value is at 7.3×10^6 years with a peak value of 4.4×10^{-6} Sv/a. The indicator is independent of the repository formation or concept and is recommended for application.

5) The indicator radionuclide concentration in the usable water near the surface represents a performance indicator in the proposed form. As discussed above the value of the indicator is limited, since most of the radionuclides, released from the CRZ are non-naturally occurring radionuclides. Therefore, it would be much more useful to consider the whole spectrum of released radionuclides and compare its radiotoxicity concentration with the radiotoxicity concentration of natural radionuclides. However, such an indicator is well known, usually denoted as radiotoxicity concentration and used in national and international studies (Becker, *et al.*, 2009). Therefore, the indicator radionuclide concentration in the usable water near the surface as proposed here is not recommended for further use.

In general, the use of complementary indicators representing safety functions helps to demonstrate the ability of the system to provide the safety functions needed. These indicators provide a way to evaluate how individual barriers (and sub-systems) behave and contribute to the overall safety concept and the performance of different repository designs (Wolf, *et al.* 2008). Additionally, they provide a means of demonstrating a comprehensive understanding of the overall repository system that cannot be gained from the calculation of dose or risk alone.

Figure 2: The cumulative released quantities from the CRZ (clay)

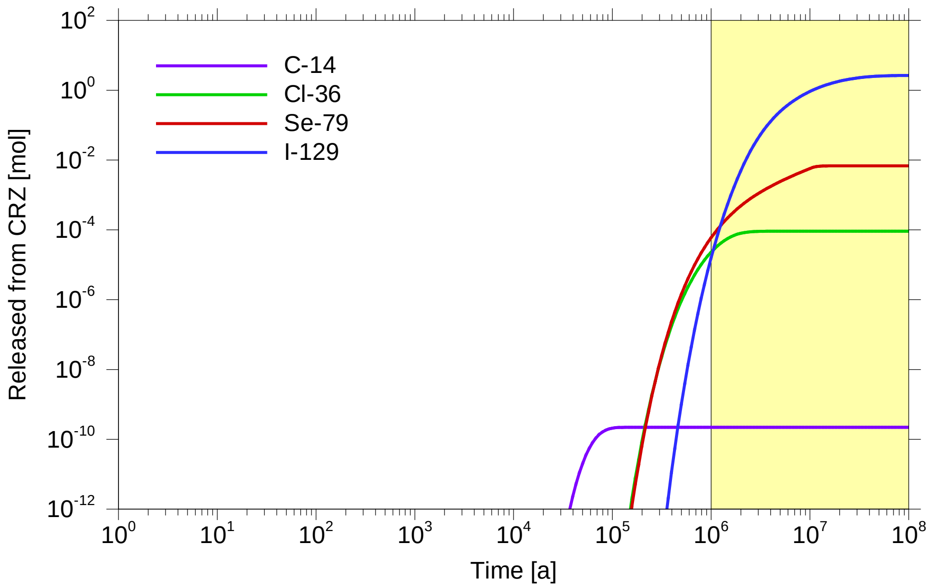


Figure 3: The contribution to radiotoxicity at the CRZ (salt)

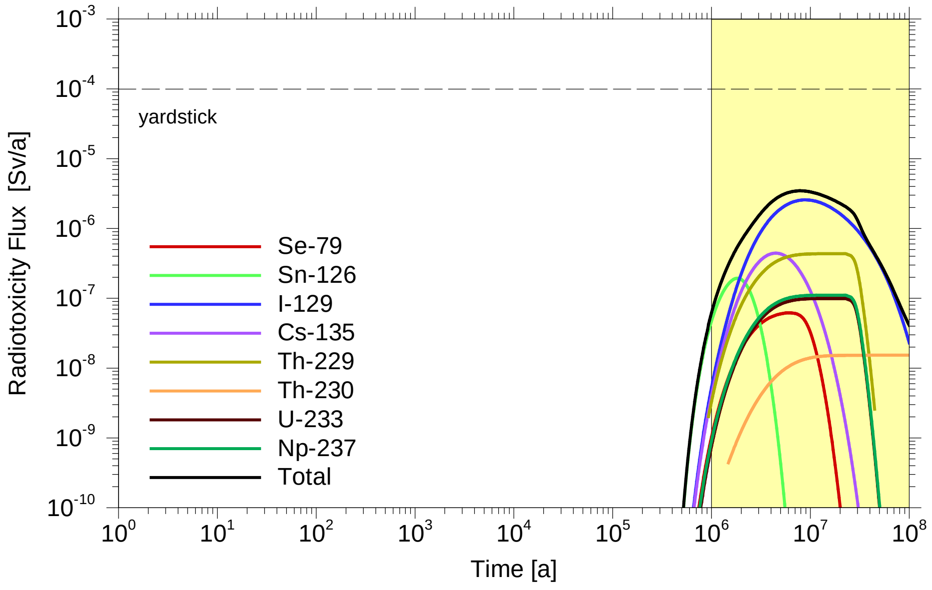
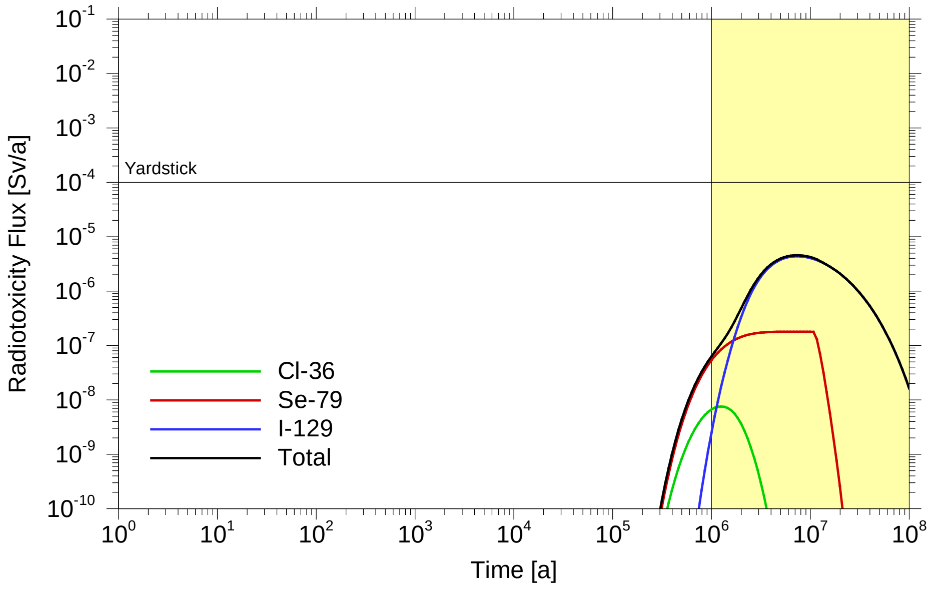


Figure 4: The contribution to radiotoxicity at the CRZ (clay)



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Study on operational safety issues in the Japanese disposal concept

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Introduction

In Japan, vitrified high-level radioactive waste (HLW) and certain types of low-level radioactive waste that results from the reprocessing of spent fuel and classified as TRU waste will be disposed of in deep geological formations.

NUMO aims to ensure the safety of local residents and workers during the operational phase and after repository closure and will therefore establish a safety case for the geological disposal programme at the end of each stage of the stepwise siting process. Although the Japanese programme is still in the stage before initiation of the siting process, updating the generic (non-site-specific) safety case is required for building confidence among stakeholders. This study focuses on operational safety issues for the Japanese HLW disposal concept.

Operational procedures

The following operational procedures after the construction of the underground facility are addressed in this study. This study focuses mainly on the operational phase, with the exception of the transport of the waste from the reprocessing facility to the repository.

In the surface facilities:

- inspection of the waste forms;
- packaging of the waste forms, e.g. into a metal overpack in case of HLW;
- transport of the overpack to the entrance of the access shaft/ramp.

In the underground facility:

- transport of the overpack in the access shaft/ramp (shaft and ramp are the alternatives for accessing the underground facility);
- emplacing the overpack in the disposal tunnel/pit;
- backfilling the remaining voids and the disposal tunnels;
- closing the access ramp and access and ventilation shafts.

Operational activities will start in a disposal panel once construction is complete; construction of the next panel will be carried out simultaneously with operations in existing panels (NUMO, 2011a).

Requirements for ensuring operational safety

Operational safety is aimed at the radiological and non-radiological protection of local residents and workers (NUMO, 2011a).

The requirements applying to radiological protection during the implementation of geological disposal are similar to those for operation in other nuclear facilities. The containment of radionuclides is ensured by the design of the facility and packaging, for example stainless steel canisters for vitrified waste and metal overpacks, and exhaust air filtering. Radiation shielding can be provided by e.g. thick concrete walls. Controlling radiation exposure to workers is also an essential component of radiological protection.

The requirements for non-radiological protection are similar to those in conventional civil engineering and mining projects. It is necessary to maintain an appropriate working environment, for example temperature and relative humidity in tunnels, and to secure evacuation routes.

Safety measures for preventing incident situations

The safety measures for preventing incident situations in the surface facilities, such as the waste inspection facility, are similar to those in other nuclear facilities, while those for the underground facility are specific to geological disposal. Incident situations are addressed based on an event-tree analysis as follows:

- dropping the waste form/overpack during handling (surface/underground);
- loss of electric power (surface/underground);
- fire (surface/underground);
- collision of transporter vehicle or deposition machine with tunnel wall (underground);
- damage to buildings (surface);
- rock fall (underground);
- explosion (surface/underground);
- flooding of tunnels (underground);
- flooding of the surface facilities (surface).

Various engineering measures will be designed based on the defence-in-depth principle to prevent the above incident situations.

For example, dropping the metal overpack during deposition in a disposal pit can be considered. This might be due to overloading of the suspension wire, probably as a result of seismic motion, or a fault in the handling device. To prevent dropping of the metal overpack, a double wire should be used for suspension or fail proof machinery should be used for the handling device. Periodic maintenance is also important. Moreover, the maximum height from the pit bottom to the deposition machine should be restricted to prevent damage to the overpack if the above measures fail.

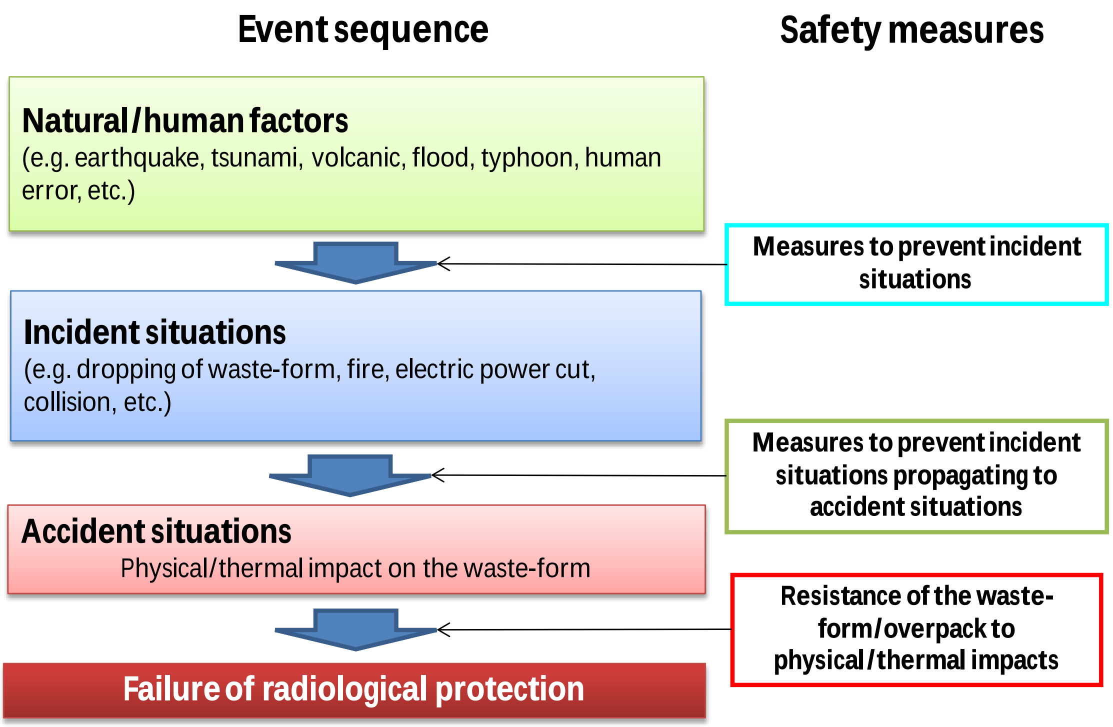
The overpack will be transported with a transporter vehicle along the access ramp (NUMO, 2011a). If the vehicle goes out of control, e.g. due to brake failure, it could be accelerated by the slope of the ramp and finally collide into a wall and/or another obstacle. To prevent this scenario, the vehicle will be equipped with an interlock device and periodic maintenance will be necessary. For the situation where the vehicle goes completely out of control, a speed-reducing run-out should be placed at a corner of the ramp. The vehicle should be equipped with impact reducing structures such as shock absorbers.

As another example, electric power may fail due to an explosion during construction of the facility in sedimentary rock. It is assumed in this case that methane gas is escaping from the surrounding rocks. During normal operation, the ventilation system can remove methane gas and the gas concentration can be monitored. If the ventilation stops for a longer period of time due to loss of electric power, the concentration of methane gas in the tunnel may increase. In this situation, the gas monitoring equipment would not function and the risk of a methane gas explosion might increase. To counter this risk, an explosion-proof design should be used e.g. for the excavation machine. In addition, if methane gas flows from the construction area to the waste deposition area, a gas explosion could occur in the deposition area. To prevent this, the ventilation system should be separated between the construction and deposition areas.

It is also possible that an earthquake could damage the tunnels, resulting in rock falling onto the waste deposition machine. Rock support will be installed as a measure for preventing rock fall. The resistance of underground tunnels to seismic wave motion has been confirmed by numerical analysis (NUMO, 2011b). As the acceleration of seismic waves underground is smaller than at the surface and the rock pressure on the tunnel wall is sufficiently high, the tunnel wall will not be damaged.

Although the measures for preventing other incidents are not mentioned here, they will be considered in the safety design of the facilities. Figure 1 illustrates the event sequence and safety measures mentioned above.

Figure 1: Schematic illustration of event sequence and safety measures



Resistance of the metal overpack to physical impacts in accident situations

Incidents can be prevented by multiple safety measures as mentioned above. Here, it is assumed that all the measures fail, resulting in an accident situation. If the metal overpack cracks due to an incident, radionuclides could be leached. In order to confirm the resistance of the metal overpack to physical impacts, numerical simulations are performed based on conservative assumptions. This study considers incidents associated with an accidental impact on the metal canister in the underground facility as follows:

- *Case I:* Dropping the overpack onto the bottom of the disposal pit during deposition.
- *Case II:* Collision of the transporter vehicle with the tunnel wall in the access ramp.
- *Case III:* Collision of the overpack with the tunnel wall due to methane gas explosion.
- *Case IV:* Rock fall on the deposition machine due to damage to the tunnel wall.

It is assumed that the overpack is damaged by crack penetration through the outer to the inner surface of overpack when the equivalent plastic strain on the overpack exceeds the strain limit of carbon steel (JIS SF340A) of 0.24.

Figure 2(a) shows the results of dropping the overpack onto the bottom of the disposal pit from a height of 5 m. The bottom of the pit was assumed to be a barely deformable solid for the conservative calculation. This calculation shows that a corner of the overpack was deformed, but crack penetration does not occur. It has been confirmed that there is no crack penetration when the overpack falls from a height less than 50 m.

For the second case, the collision of the overpack with the tunnel wall in the access ramp has also been simulated (NUMO, 2011a). In this calculation, it is assumed that the transporter vehicle is out of control on the access ramp, reaches a speed of 35 km/h and collides with the tunnel wall. Although the transport container and vehicle may reduce the impact of the collision during actual incidents, it is assumed here for the conservative calculation that the overpack is unprotected and collides directly with the wall. However, the equivalent plastic strain was a maximum of 0.046 at the surface and crack penetration did not occur (NUMO, 2011a).

Figure 2(b) shows the result for the collision of the overpack with the tunnel wall following a methane gas explosion. In this calculation, it is assumed that methane gas flows from the construction area to the deposition tunnel, explodes and the blast blows over the overpack being deposited. The overpack flies through the air with a maximum speed of 72 km/h and collides with the tunnel wall. This calculation also shows that a corner of overpack was deformed, but that crack penetration does not occur.

Figure 2(c) shows the result of a rock fall onto the overpack. In this calculation, the rock mass with a cubic form of 5 m ´ 5 m ´ 5 m (around 200 tonnes) falls directly from a height of three metres onto the unprotected overpack for the conservative calculation. However, only slight damage was observed on the surface of overpack, while brittle deformation of the rock occurs.

The overpack is thus highly resistant to physical impact and no accidents will occur.

Recovery from accident situations

The numerical calculation of the physical impacts on the overpack showed no leaching of radionuclides during accident situations. To maintain a sustainable disposal programme, techniques for recovering from incidents should be prepared. Figure 3 shows a schematic illustration of the procedure for recovery from rock fall (Case IV above). The rock fragments can be removed by the existing machines and the damaged roof will be reinforced. After removing the deposition machine, the overpack can be removed, for example by overcoring of the bentonite buffer. To reduce the radiation exposure of workers, these machines should be remotely handled. These techniques should be developed before starting operation.

Figure 2: Distribution of equivalent plastic strain on the cross-section of the overpack after incidents

Red areas indicate strain exceeding the limit; arrows indicate the direction of the stress

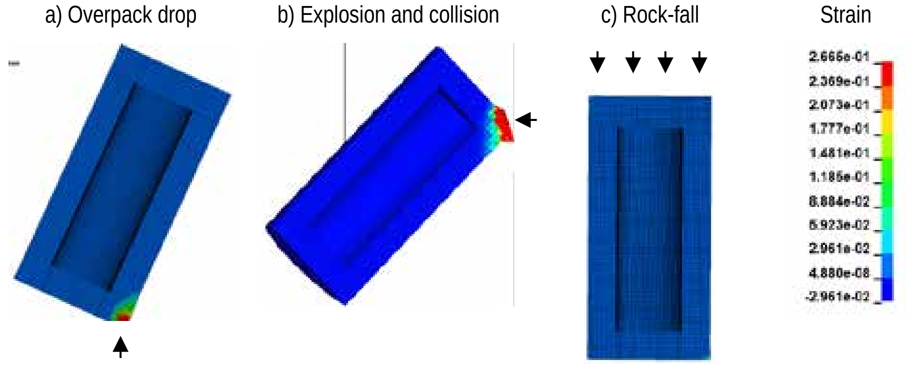
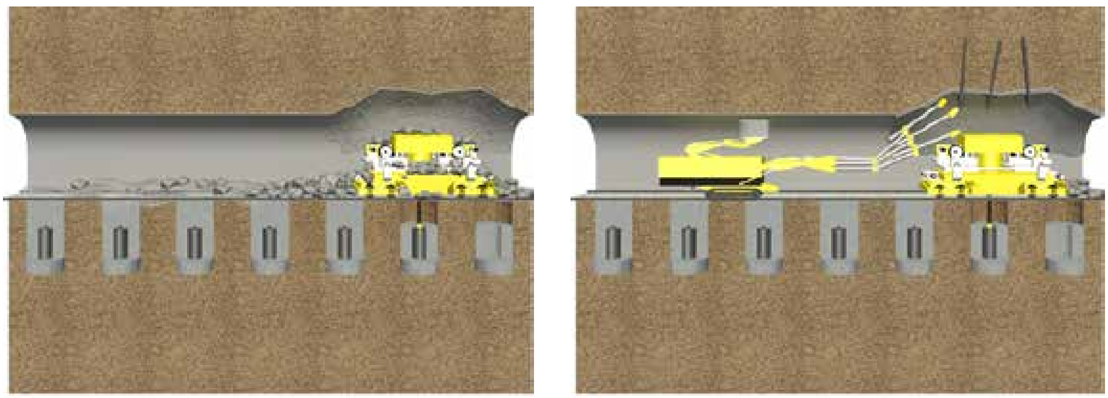


Figure 3: Conceptual view of the recovery technique from a rock fall incident during the deposition of an overpack into the disposal pit

Left – rock fall incident during the deposition process
Right – removing the rock and reinforcing the damaged roof



Conclusions

The operational safety of the Japanese disposal concept can be assured by applying requirements similar to those for other nuclear facilities and conventional civil engineering projects. To prevent incidents, safety measures can be designed based on the defence-in-depth principle. The metal overpack has a high resistance to physical impacts resulting from incident scenarios. Although the risk of radionuclide leaching is negligible, remotely controlled recovery techniques should be developed before starting operation.

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Operational safety of geological disposal: IRSN project "EXREV" for developing a safety assessment strategy for the operation and reversibility of a geological repository

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Context

A high-level waste geological disposal facility is envisioned by the legislator in the French Planning Act n°2006-739 of 28 June 2006. This act sets major milestones for the operator (Andra) in 2013 (public debate), 2015 (licensing) and 2025 (operation). In the framework of the regulatory review process, IRSN's mission is to conduct an assessment of the safety case provided by Andra at every stage of the process for the French regulator, namely the Nuclear Safety Authority (ASN).

In 2005, IRSN gathered more than twenty years of research and expertise in order to provide a comprehensive appraisal of the "Dossier 2005" prepared by Andra, related to the feasibility of a geological disposal in the Callovo-Oxfordian clay formation. At this time, the description of the operational phase was only at a preliminary stage, but this step paved the way for developing an assessment strategy of the operational phase. In this perspective, IRSN set up the EXREV project in 2008 in order to build up a doctrine and to identify key safety issues to be dealt with.

Objectives

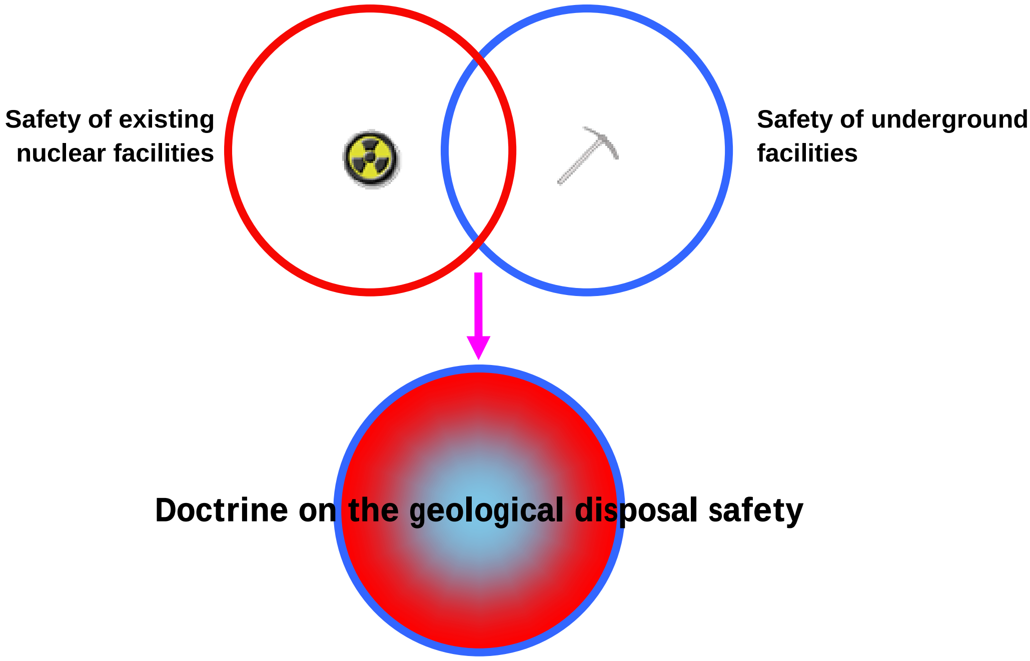
This doctrine had to take into account constraints that come from the specific concept of such a facility, which combines safety issues derived from classic nuclear facilities with those from construction and operation of conventional underground installations such as mines or tunnels. In any case, the safety approach that shall be developed must remain coherent with "classic" nuclear risks and the two main safety functions associated with the components of a nuclear facility: containment and radiation protection. The safety approach must also address in particular the interactions between underground and surface areas where industrial processes are to be deployed, the management of operating and construction activities performed in parallel, and finally the management of "reversibility". The latter could comprise the following aspects:

- the ability to retrieve the waste packages;
- the possible re-design of the waste disposal process;
- re-engineering of parts of the facility and their components.

The EXREV project then aimed at gathering and possibly "merging" knowledge and studies from the nuclear safety world with experience from the operation of deep underground facilities, such as mines, tunnels, shafts, drifts, etc. Both worlds have in common an extensive feedback from existing and past facilities, and both worlds have safety and security approaches considered as “best practices”. The challenge is to take benefit of these practices and adapt or possibly develop new practices, thus ensuring the highest level of safety possible for this new kind of underground nuclear facility. For instance, it is well-known that nuclear facilities can be viewed as vast, extremely clean, well-lit areas, while mines are often perceived as cramped, dust-filled and somewhat dark spaces. The same kind of comparison can be made in several fields, such as that of ventilation systems (static nuclear ventilation systems versus dynamic *ad hoc* comfort-driven ventilation systems).

The IRSN preferred approach for developing such a safety assessment strategy, displayed schematically in Figure 1, is to underline the basic principles that must be taken into account when operating a nuclear facility and an underground facility.

Figure 1: The IRSN preferred approach for developing a safety assessment strategy



Partners

Operators of national and international underground facilities (Modane-Bardonecchia Frejus Tunnel,...) participated in the project, and experts from the mining sector (Mines Paris Tech Centre of Geosciences) contributed in identifying the above-mentioned principles. Feedback from international experience was also considered during the EXREV project (former Yucca Mountain project, US WIPP, South African deep mines, Canadian uranium mines,...).

EXREV was also linked with the IAEA GEOSAF project on international harmonisation of approaches in the construction and evaluation of the safety case for a geological disposal and the subsequent GEOSAF 2 project on the integration of an operational phase and post-closure phase into the safety case.

General findings and main conclusions for the regulatory review

- In terms of methodology of regulatory review in the context of geological disposal, existing methods such as the safety evaluation performed by IRSN on various nuclear facilities remains valid.
- However, the parameters associated with the characterisation of the considered risks (fire, flood,…) need to take into account the peculiarities of such a facility.
- Furthermore, some specific risks or situations need to be addressed without any substantial feedback experience from the operation of existing nuclear facilities (concomitant activities, evacuation in the case of fire,…).
- Finally, the identification of limits, controls and conditions for the operational phase remains a challenge, since it has to integrate the dimension of long-term safety. The numerous links between pre- and post-closure arguments of the safety case call for a methodology to verify continuously that the operator is always on the right track to achieving its target, namely the conditions of the repository at the time of closure which form the basis of the demonstration that the facility is safe over the long term.

These four findings lead to identifying five conclusions for the preparation of the regulatory review:

- The analysis of the design, the maintenance, the coherence between the provisions adopted and the considered risks (especially those that are specific to a geological disposal) should be deepened. This underlines the questions related to the technologies used in the facility, the architecture, the components' robustness and easy maintenance, as well as a deep understanding of the mechanisms associated to the ageing of the above-mentioned components.
- The analysis of the scenarios used by the licensee, especially those that are bounding, should be carried out with a good understanding of the peculiar characteristics of a geological disposal.
- Risks associated with activities running in parallel over extensive periods of time should be considered as essential.
- An analysis of the adequacy of a monitoring and surveillance programme during the operational phase, which would consider several objectives, should be performed as well.
- As stated above, a deeper knowledge of the various situations and parameters that influence the "initial state" of the closed repository (namely the characterisation of the set of parameters that control the post-closure safety assessment) should be sought as well.

Perspectives

These conclusions paved the way for future IRSN research. As of 2013, studies and R&D have begun on the following topics:

- Fire risks:
  - characterisation of fires in underground spaces;
  - thermal response of ILW emplacement cells on temperature rise aggressions;
  - quantification of effects of fire on specific target in confined environment;
  - integration of different confined environment in IRSN's simulation tools.

- Handling risks:
  - characterisation of situations where waste packages transfer operations in the disposal facility are stopped/stalled(stalling cinematic chain of transfer);
  - consequences of these situations on safety-relevant systems and on the general level of risks in the facility.
- Activities performed in parallel (co-activity):
  - methods (including in other industries) for organising safely activities performed in parallel;
  - definition of situations (such as evacuation in the case of fire in the underground area) that should be taken into account in the analysis of these risks.

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Session 7.3

The Broader Perspective: Safety Case, Programme Decision and Society

Independent modelling in SSM's licensing review of a spent nuclear fuel repository

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Background

In 2011 the Swedish Nuclear Fuel & Waste Management Co. (SKB) submitted a license application for construction of a geological repository for spent nuclear fuel at Forsmark. SKB's disposal method, the KBS-3 method, involves disposing of the spent nuclear fuel in cast iron canisters with an outer layer of 5 cm copper. The canisters will be placed in vertical deposition holes at approximately 500 m depths in crystalline bedrock. Each canister is surrounded by a buffer of swelling bentonite clay. The repository is designed to accommodate 6 000 canisters, corresponding to 12 000 tonnes of spent nuclear fuel.

The license application is supported by a post-closure safety assessment, SR-Site (SKB, 2011). Along with other parts of the application, SR-Site is currently being reviewed by the Swedish Radiation Safety Authority (SSM). The main method for review of SKB's licensing documentation is document review carried out by SSM, supported by SSM's external experts. However, SSM's document review is also supported by regulatory modelling, technical reviews of SKB's quality assurance programme and consideration of external review comments partly from two broad national consultations and an international peer review organised by the OECD's Nuclear Energy Agency (NEA, 2012). SSM's review is divided into three main phases: the initial review phase, the main review phase and the reporting phase (Dverstorp, *et al.*, 2011). The overall goal of the initial review phase is to achieve a broad coverage of SR-Site and its supporting references and in particular to identify the need for complementary information and clarifications to be provided by SKB, as well as to identify critical review issues that require a more comprehensive treatment in the main review phase. SSM completed the initial review phase at the end of 2012.

Developing a regulatory modelling capacity

SSM and its predecessors have, for several decades, been developing independent models to support regulatory reviews. Modelling teams have been established, combining both in-house and external expertise. SSM's independent modelling can be referred to one of the following three categories:

- use of simple scoping calculations;
- use of SKB's own models (with other equation solvers);
- use of alternative conceptual models.

Simple scoping calculations may be used to verify if SKB's modelling results are reasonable and to check the impact of individual process descriptions or parameters on radiological consequences. By replicating SKB's calculations, using their own models or SSM's interpretations thereof, SSM can gain insight into the details of SKB's calculations that cannot be attained simply by reviewing SKB's modelling reports. Finally, the use of alternative conceptual models provides a means to explore different types of uncertainty related to safety critical review issues.

The experience from previous reviews of SKB's preliminary safety assessments, e.g. SR-97 (SKI, 2000) and SR-Can (Xu, *et al.*, 2008), illustrates that independent modelling is an effective tool for checking the quality and transparency of SKB's safety case. However, because independent modelling is a labour intense and time consuming activity, efforts must be restricted and cannot cover all modelling work presented in SR-Site.

Initial review of SR-Site

In the following we focus on SSM's independent modelling of SKB's consequence calculations (radionuclide release, geosphere transport and dose calculations) in SR-Site. At this early stage of the licensing review, the primary objective is to identify critical review issues as mentioned above. Independent modelling will also be used to address process modelling supporting SKB's safety case, but this will take place in the main review phase when SSM has a better picture of what are safety critical issues.

SKB's approach to consequence analyses

Because the KBS-3 method relies heavily on the containment safety function, canister integrity is on the focus of the safety case. In SR-Site, there are two scenarios for which canister failures are not excluded, namely the scenarios "canister failure due to corrosion" and "canister failure due to shear load" (SKB, 2011). Hereafter these two scenarios are referred to as "corrosion scenario" and "shear load scenario". The consequence analyses of these scenarios are based on models which describe radionuclide transport in the near field, far field and biosphere. Radionuclide transport in the near field is modelled with the compartment model COMP23 (Cliffe and Kelly, 2006; SKB, 2010) that models processes related to radionuclide release and transport in the canister interior, the bentonite buffer and in the deposition tunnel backfill. SKB's far field transport is modelled with FARF31, a one-dimensional advection-dispersion model with matrix diffusion and sorption to describe groundwater radionuclide transport in fractured rock (Norman and Kjellbert, 1990; SKB, 2010). Doses to a representative individual in the most exposed group are obtained by multiplying the calculated flux from the geosphere to the biosphere with landscape dose conversion factors (LDF). The LFD are calculated with a complex model, the so-called Radionuclide Model (Avila, *et al.*, 2010) that describes the continuous development in time of both terrestrial and aquatic biosphere objects. The LDF are derived for a constant unit release to the biosphere, separate from the geosphere transport calculations.

SSM's independent modelling

SSM's independent modelling in the initial review phase is an interpretation of the models that SKB utilises in SR-Site and the numerical software Ecolego (2011) is used. Ecolego is a compartmental modelling software in which the COMP32 near-field transport model and the Radionuclide Model for dose assessment is implemented. The discretisation method proposed by Broed and Xu (2008) is used to implement the FARF31 model in Ecolego. Deterministic and probabilistic radionuclide release and transport calculations for SKB's corrosion and shear load scenarios, including LDF values, are reproduced.

In parallel, independent modelling was performed by SSM's consultants. Pensado and Mohanty (2012) used an alternative compartmental representation of SKB's nearfield and far-field release and transport models to evaluate SKB's results for the corrosion and shear load scenarios.

Some examples of results

Even if SSM has identified some problems regarding clarity in SKB's model descriptions and traceability of the input data used for the calculations, it was possible to reproduce SKB's calculations with certain assumptions. Using the models and input data adopted by SKB, the reproduced results from the two scenarios are comparable with SKB's results. However, to draw firm conclusions, a number of issues need to be clarified and further investigated in the main review phase. In the following we present some examples of how SSM's independent modelling has contributed to the review goals of the initial review phase.

Need for complementary information

A general description of SKB's radiological exposure assessment is given in Avila, *et al.* (2010). However, in the process of reproducing LDF values SSM identified a lack of justification for a number of assumptions. There is no detailed description about how exposure pathways were selected, i.e. justification of the selected exposure pathways is insufficient. For instance the combustion of peat for energy production is known as an important route of exposure in previous preliminary safety analysis (Bergström, *et al.*, 1999), but it is not included in SR-Site.

Traceability of data

SKB's model description for radionuclide transport in the near field has improved since SR-Can; for instance, a concrete example of numerical value of transfer rate between compartments is explicitly given in SR-Site. However, the information provided regarding the model and the input data is still not always clear. In the cases of canister failure due to shear load and the growing pin-hole the transport resistances are given as zero or negligible (see Table G-7 and G-4 in SKB, 2010) with no quantifying information. This makes reproduction of SKB's calculations difficult. Since the radionuclide transport in near field is modelled by a compartmental model the mass transfer is modelled by a transfer rate, which is described as the inverse of the transport resistance. If the parameter value used for the resistance is zero it means the transfer rate goes to infinity. If only he parameter value is stated as negligible we do not know what SKB considers as negligible. Clarification of this matter is needed.

Identified conceptual uncertainty for further review

Pulse releases are defined as a sudden release of certain mobile radionuclides (instant release fraction) at the time of canister failure. SKB concludes that pulse releases give negligible contributions to the probabilistically calculated mean dose with the argument: "*The width of the dose curves in the biosphere is typically 1 000 years. The likelihood that an exposure due to a pulse release, p_{Expo}, is present at a given point in time during the 100 000 year interval is thus $10^{-2} \cdot p_{Fail}$. (The likelihood of overlaps between pulses is very small due to the low probabilities).*" However, a simple scoping calculation, taking ^{129}I as an example, shows that the amount of pulse release due to IRF is about 30% of the total release for the case of canister failure due to corrosion with a normal dissolution rate. Therefore, SSM will elaborate further on the issue of pulse releases in the coming main review phase. Independent modelling will be carried out to analyse pulse releases in an integrated system of geosphere and biosphere models, rather than the decoupled system with separate analyses of pulse releases carried out by SKB.

Discussion

During the initial review phase SSM has identified a number of issues requiring either clarifications, complementary information from SKB or further in-depth review by SSM. Important issues include the inconsistency between the documents and the actual modelling performed in SKB's dose assessment, QA problems in the consequence analyses, and insufficient justification of assumptions. We can already now conclude that independent regulatory modelling combined with traditional document review is an effective way to enhance the authority's licensing review process.

SSM communicated the results from the initial review phase on the 29 October 2012 by handing in a written statement to the Land and Environment Court and SKB. SKB has also been informed of the results of SSM's independent modelling through SSM's request for complementary information.

In the main review phase, to follow, we plan to continue with reproduction of selected "What if?" and "residual" scenarios as well as "barrier function" scenarios that were not covered in the initial review phase. Moreover, to further investigate uncertainty in the calculated LDF values we plan to perform some alternative modelling, i.e. simpler "reference biosphere" modelling with the objective of exploring the uncertainty of various properties of SKB's biosphere objects, including their evolution in time.

Further, the details of the derivations of the probabilities of canister failure scenarios, which are beyond scope of this paper, will be handled in the in-depth review of the main review phase.

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Setting up a safe deep repository for long-lived HLW and ILW in Russia: Current state of the works

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The concept of RW disposal in Russia in accordance with the Federal Law "On Radioactive Waste Management and Amendments to Specific Legal Acts of the Russian Federation" No. 190-FL dated 11 July 2011, is oriented at the *ultimate* disposal of waste, without an intent for their subsequent retrieval.

The law 190-FL has it as follows:

- A radioactive waste repository is a radioactive waste storage facility intended for disposal of the radioactive wastes without an intent for their subsequent retrieval.
- Disposal of solid long-lived high-level waste and solid long-lived intermediate-level waste is carried out in deep repositories for radioactive waste.
- Import into the Russian Federation of radioactive waste for the purpose of its storage, processing and disposal, except for spent sealed sources of ionising radiation originating from the Russian Federation, is prohibited.

For safe final disposal of long-lived HLW and ILW, it is planned to construct a deep repository for radioactive waste (DRRW) in a low-pervious monolith rock massif in the Krasnoyarsk region in the production territory of the Mining and Chemical Combine (FSUE "Gorno-khimicheskiy kombinat").

According to the IAEA recommendations and in line with the international experience in feasibility studies for setting up of HLW and SNF underground disposal facilities, the first mandatory step is the construction of an underground research laboratory.

An underground laboratory serves the following purposes:

- itemised research into the characteristics of enclosing rock mass, with verification of massive material suitability for safe disposal of long-lived HLW and ILW;
- research into and verification of the isolating properties of an engineering barrier system;
- development of engineering solutions and transportation and process flow schemes for construction and running of a future RW ultimate isolation facility.

Radioactive waste final disposal facility; construction purpose and basic indices

Long-lived and medium level RW disposal facility construction purpose

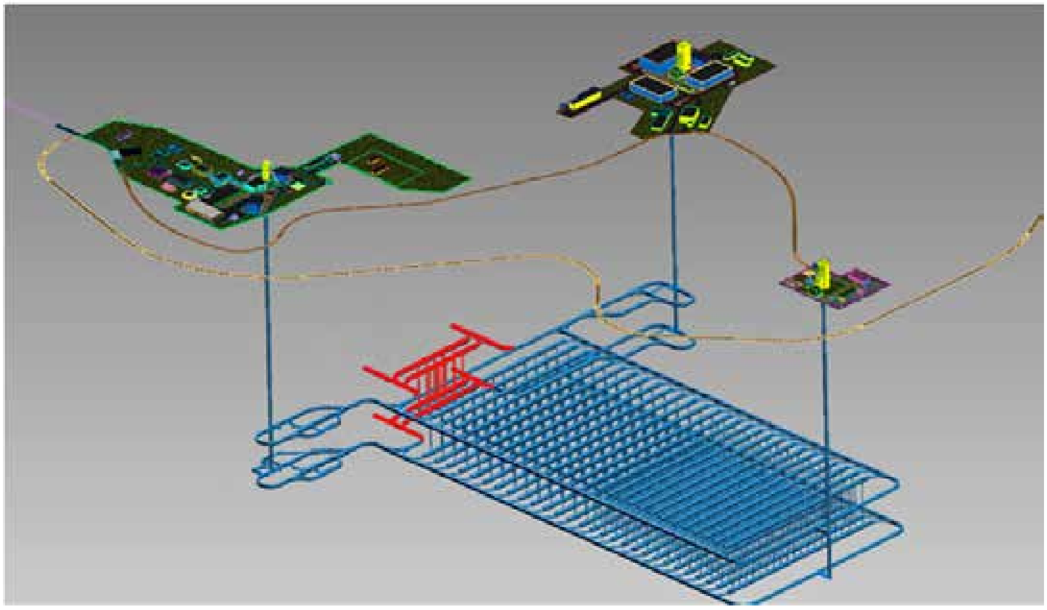
The purpose of the facility is to create an environmentally safe, technically reliable and economically acceptable disposal facility for long-lived and medium-level RW in deep geological formations, including future SNF recycling at MCC.

Basic indices

- Facility location – Krasnoyarsk region, 4.5 km from the Yenisei River, 6 kilometres from the town of Zheleznogorsk.
- The depth of the underground facilities – 450-525 m.
- Capacity of the RW:
  – vitrified long-lived HLW – 4.5 thousand m^3 (7 500 casks);
  – long-lived ILW and HLW with insignificant heat liberation – 155 thousand m^3.
- The term of the facility full loading – 2047.

The scheme of RW final disposal facility and underground laboratory is displayed in Figure 1. Underground constructions of the underground laboratory are highlighted in red.

Figure 1: Schema of RW final disposal facility and underground laboratory



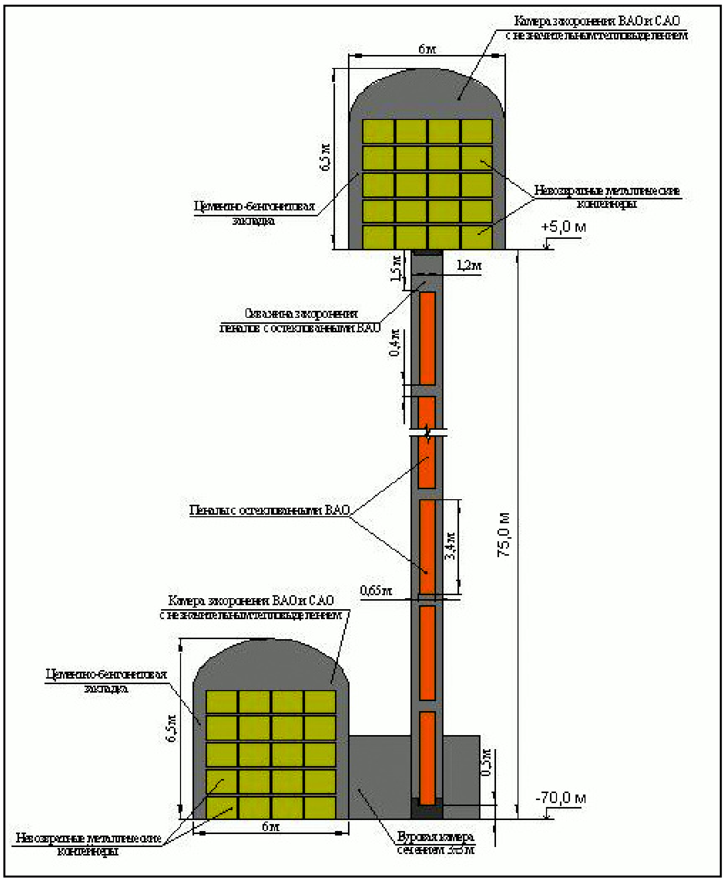
Long-lived ILW and HLW with insignificant heat liberation will be placed in extended cameras for RW disposal, located on the up (450 m depth) and down (525 m depth) horizons. Containers with RW will be placed in piles in the sections of the disposal cameras.

Vitrified long-lived HLW with high heat liberation will be placed in vertical holes 1.2 m in diameter and 75 m deep, located between horizons (Figure 2).

A safe ultimate isolation of long-lived solidified HLW and ILW in deep geological formations will enable to:

- spare future generations the burden of having to handle accumulated long-lived RW;
- substantially improve the ecological situation on production sites of “PO “Mayak”, MCC and SCC combined;
- avoid long-standing operational expenditure on storage of amassed long-lived HLW and ILW in temporary surface and near-surface facilities;
- ensure looping of a nuclear fuel cycle: safe disposal of conditioned HLW and ILW fractions from future SNF recycling at MCC.

Figure 2: Schema of vitrified long-lived HLW storage



RW final disposal facility construction current state and perspective plans

- RW final disposal facility construction preliminary project documentation was developed and confirmed: "Declaration of Intention" (2008) and "Feasibility Study" RW final disposal facility construction (2011).
- Positive expert report by the Federal Agency for Subsoil Usage on the location applicability for RW final disposal facility construction was received (2012).
- Regional public hearings approval on environmental impact assessment was conducted (2012).
- At the present moment design and survey work on the "Project Documentation" stage is in operation, the deadline is 2015.
- Beginning of long-lived ILW and HLW final disposal facility and underground laboratory construction is planned for 2016.

Construction plans and content of research in the underground laboratory

On completion of the construction of auxiliary and process shafts and shaft insets on horizons, the works of underground laboratory construction are envisaged: two mines working on the upper and the lower horizon each, one above the other, and five vertical holes between the horizons; six deep flat exploration holes on the upper horizon.

The underground laboratory will engage in experimental try out of the operations of RW deposition chambers and hole construction, mastery of process operations of RW handling and setting up of an engineering barrier system. An integrated investigation into the isolating properties of engineering barriers will be carried out; the dynamics of thermal processes in boreholes and massive material will be investigated as well.

- *Phase I – 2016-2018 (3 years):* Research of formation material during underground research laboratory vertical and horizontal underground facilities construction.
- *Phase II – 2019-2021 (3 years):* Research of formation material in underground facilities and exploration holes. Try out of the operations of RW deposition chambers and holes construction.
- *Phase III – from 2022:* Mastery of process operations of RW handling.

Carrying out of a complex of field observations and laboratory experiments in the permanent in-mine structures of the underground research laboratory

The content of planned research will be in compliance with the recommendations of SRC "Rosnedra" based on the results of expert review of the materials of geological investigations conducted on the site, IAEA recommendations and similar research done internationally in the existing underground laboratories.

Carrying out the complex of engineering and geological surveys, field observations and laboratory experiments in the permanent in-mine structures, immediately at the arrangement depth of future deposition chambers of conditioned radioactive waste will ensure reliable assessment of geological, hydrogeological, structural and tectonic, and seismic conditions of the rock mass under investigation, physical and mechanical, thermophysical, and sorption-migrational characteristics of formation.

Selection and justification of suitability of a site for URL and DRRW construction

The explorations conducted in the Nizhnekansky massif area with the aim of selecting a prospective site for construction of an RW ultimate isolation facility started in the early 90s, involving experts of the RF Minatom, Russian Academy of Sciences, and geological organisations of the Krasnoyarsk region.

In 1992-2001, areal studies were carried out, which resulted in selection of two prospective sites.

In 2002-2005, integrated geophysical exploration works were performed on the more promising of the earlier selected sites – "Yeniseisky", located within the limits of Zheleznogorsk town Closed Administrative Territorial Formation (CATF). The explorations were conducted stepwise, with successive narrowing of surface area and increasing concentration of investigation points, from the initial 70 km^2 to more detailed research at an area of 25 km^2.

Based on the results of all exploratory works done from 1993 through 2005, a site was singled out within the limits of the investigated area in the "Yeniseisky" sector and recommended to be subsequently used for carrying out detailed geological engineering surveys with the prospect of constructing a facility for ultimate isolation of HLW and ILW containing long-lived radionuclides.

The recommended site is located 4.5 km from the Yenisei River, 6 kilometres from the town of Zheleznogorsk.

In 2008-2011, integrated geophysical exploration works were carried out in the recommended area, including surface surveys, mining and geological, and hydrogeological investigations with the use of exploration holes.

Based on the results of conducted engineering and geological surveys with full core sampling and a full-scale complex of geophysical and hydrogeological investigations, seismic surveys from the surface, research into the stress and strain state of natural condition massive material the following rock mass characteristics to a depth of 700 metres in the area of facility construction site were determined, providing massif suitability for RW disposal:

- Structural and tectonic characteristics of the rock mass are favourable for construction of a long-lived RW disposal facility. The massive material is characterised by a stable tectonic regime.
- The massive material relates to a category of medium to high strength.
- The revealed jointing zones are flat-dipping, directed downwards and *away from the Yenisei,* filled with carbon-bearing, feldspathic material, quartz or clayey materials and *are not water-transmitting,* which is proved by the results of experimental filtration research.
- Underground structures of the facility will be located below the local drainage base level, i.e. the Yenisei River bed, which rules out the possibility of groundwater ingress into the surface water bodies. At the same time, permeability coefficient values in the discovered range of depths are very low, as is characteristic of aquitards.

Based on the packaged investigation results submitted to expert reviewing, a positive expert opinion was issued in April 2012 by the Federal State Institution “State Reserves Commission” (SRC “Rosnedra”) for the “evaluation stage”, with an emphasis *on site suitability for construction of a facility for ultimate isolation of solidified RW*.

Preliminary assessment of DRRW construction safety

For preliminary assessment of DRRW safety, mathematical modelling techniques were used.

Modelling of sorption-migrational processes has demonstrated that the ecological safety of long-lived RW disposal is ensured with a considerable margin. Taking into account the downward character of groundwater infiltration at the “Yeniseisky” site and favourable sorption-migrational characteristics of formation, radionuclides will not escape to the environment throughout the entire period of potential ecological hazard.

The rock mass at the DRRW site is characterised by a stable tectonic regime, such that no significant changes in the massif characteristics should be expected in the next millions of years and the ecological safety of the facility is guaranteed.

Modelling of the stress and strain state of border zones of the underground structures has demonstrated the robustness of workings. Supports will only be required in the areas of transport working intersections with the horizontal RW deposition chambers.

Definition of the OPERA safety case for radioactive waste disposal in the Netherlands

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This paper first gives a short introduction on OPERA, the current Dutch five-year research programme on disposal of radioactive waste. It then zooms in on OPERA WP (Work Package) 2 Safety Case – the OSCAR project, and presents (preliminary) results on the structure of the OPERA safety case, the subject of safety statements, and the OPERA safety assessment methodology.

Introduction

Radioactive waste policy in the Netherlands states that all kinds and categories of radioactive waste are managed by one central waste management organisation (COVRA[1]) and stored for at least 100 years at one site, above ground in engineered structures. This allows retrieval at all times. The period of interim surface storage is to be followed by geological disposal for all waste categories (low-, intermediate- and high-level waste) in one single repository. During the interim storage, disposal is to be prepared for socially, technically and economically, such that it can be implemented efficiently thereafter.

Implementing and operating a small repository is costly, in particular for countries with small nuclear power programmes such as the Netherlands. The economy of scale will force them either to implement long-term storage and wait for decades, and/or to share a repository with others. With only 525 MWe of installed nuclear capacity in the Netherlands, two research reactors, an enrichment facility and about 200 producers of institutional waste, there is no need for disposal in the short term. The volume of all categories of radioactive waste generated over 30 years is only a few thousand m^3: 60 m^3 of HLW, 10 000 m^3 of LILW and another 10 000 m^3 of NORM waste. The resulting disposal costs per m^3 are very high as long as the accumulated amount of waste is small.

The additional costs of prolonged interim surface storage are relatively small and the small volume of waste can easily be controlled in surface structures. This "interim" storage provides time both to accumulate the volume of waste and to let the amount of money, needed for disposal, grow in a capital growth fund. In 100 years' time, growth by about a factor of 10 can be obtained with a real interest rate of 2.3%. Moreover, an international or regional solution may become available during these years.

1. Centrale Organisatie Voor Radioactief Afval (COVRA) is the central organisation in the Netherlands for collection, treatment, storage and eventual disposal of radioactive wastes.

The long-term storage is in full operation now and necessary provisions for the next step have been taken as well. The capital growth fund to finance the final disposal exists; waste generators pay for this and there is a clear choice for the ownership of the waste: all liabilities are transferred to the waste management organisation COVRA. The challenge was, however, to restart the research on geological disposal after a period of almost 10 years without a national research programme.

This paper outlines the resumption, design and objectives of the present Dutch national research programme OPERA.[2] It then focuses on the results obtained so far in the OSCAR[3] project, the aim of which is to develop the structure for the OPERA safety case.

The OPERA research programme

The Dutch policy on radioactive waste management is based on a report presented to parliament by the government in 1984 (VROM). That report covered two items, viz. the long-term interim storage – at least 100 years – of all radioactive waste generated in the Netherlands, and the government research strategy for geological disposal of the waste. The report led to the establishment of COVRA in the municipality of Borsele, and the launch of subsequent research programmes on the geological disposal of radioactive waste: OPLA, CORA and now OPERA.

Structure and management of the OPERA research programme

To restart the research on geological disposal COVRA and NRG[4] together drafted an outline for a five-year research programme. Based on the recommendations of the previous programme, CORA, consultation with research institutes in the Netherlands and a workshop (ministries, universities, research institutes, nuclear organisations, ONDRAF/NIRAS and COVRA), the proposal for the research programme gained broad support among stakeholders in the Netherlands. The starting points for defining the research programme were: resolving outstanding issues from previous programmes; developing and preserving expertise and knowledge; and being prepared for site selection in case of any change to the current timetable, arising by way of future European directives, for example. The research programme started in June 2011, almost simultaneously with the Directive, which also requires research activities as a way to obtain, maintain and develop the necessary expertise and skills for the management of radioactive waste.

The research programme is called OPERA and is financed by the Dutch government (50%) and the nuclear sector (50%). Radioactive waste organisation COVRA manages the programme, collects and integrates the results, but does not carry out the research itself. It was decided to clearly separate the tasks of carrying out the research from the task of managing the programme. The research projects in OPERA are carried out by almost 20 research institutes in the Netherlands and abroad.

OPERA will detail a first roadmap for the long-term research on geological disposal of radioactive waste in the Netherlands. This roadmap will be based initially on a re-evaluation of existing safety and feasibility studies conducted more than ten years ago, making use of present international and, wherever possible, national knowledge. The focus of the programme will be on Boom Clay, as salt formations received more attention in the two previous Dutch research programmes. In particular it was decided to build on the ONDRAF/NIRAS disposal concept and research on Boom Clay. Since 1974, the Belgian

2. OnderzoeksProgramma naar Eindberging van Radioactief Afval (OPERA) – Research Programme into the Geological Disposal of Radioactive Waste.
3. OPERA Definition of the Safety Case for Radioactive Waste Disposal (OSCAR).
4. Nuclear Research and Consultancy Group (NRG), the Netherlands.

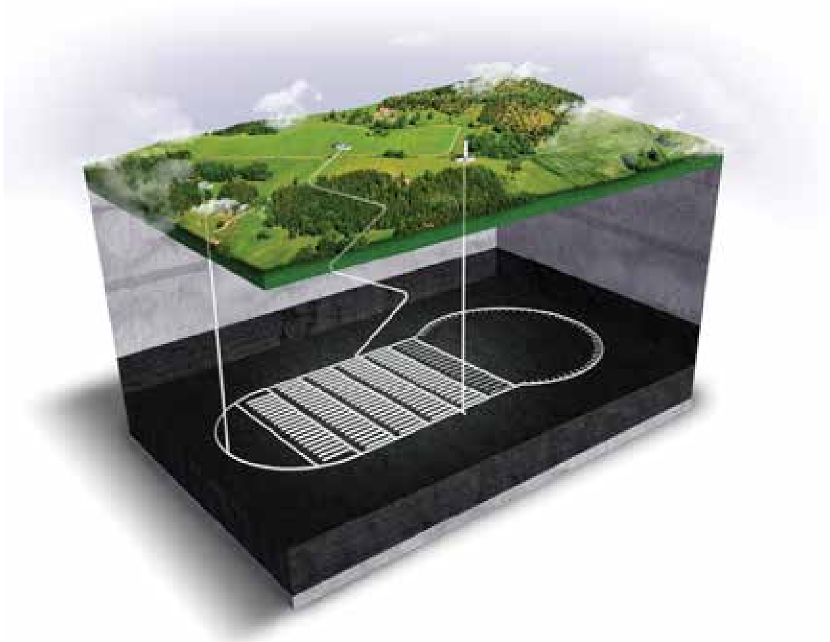
programme has developed extensive knowledge of disposal of radioactive waste in Boom Clay. The Belgian programme includes an underground research laboratory at Mol where experiments have been and still are performed to validate models.

A further aim of the OPERA programme is to build and maintain the knowledge and competence to run a geological disposal research programme and manage the interfaces between all steps of the radioactive waste management process from generation to disposal. Objectives are to reactivate research on geological disposal and to involve a broad group of (new) researchers in the field, as well as to provide access to previous research on geological disposal in the Netherlands, communicate transparently about the results and embed the (developed) knowledge in an academic curriculum. OPERA results and reports will be published at the COVRA website (ww.covra.nl).

OPERA disposal concept

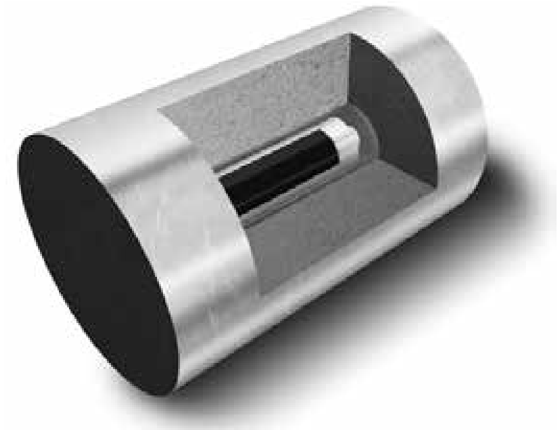
To focus the research in OPERA, a reference disposal concept was developed. Figure 1 presents an artist's impression of the OPERA disposal concept in Boom Clay (Verhoef, 2011, p. 10). The OPERA disposal facility consists of both surface and underground facilities. The underground facilities contain separate disposal sections for the different types of wastes, a pilot facility and a workshop for maintenance work, all connected by the main gallery. The main gallery is an orbicular structure, which connects with the ground level via two access shafts and/or an (optional) inclined ramp.

Figure 1: Artist's impression of a geological repository for the disposal of radioactive waste in Boom Clay



The OPERA supercontainer is based on the Belgian supercontainer concept, which consists of a carbon steel overpack, a concrete buffer and stainless steel envelope and can hold two HLW canisters or one SF canister (Humbeeck, *et al.*, 2007). In OPERA a uniform supercontainer is used for the heat-generating HLW, spent fuel from research reactors as well as the non-heat-generating HLW. Figure 2 shows an artist's impression of the OPERA supercontainer for heat-generating HLW Alternatively, a supercontainer without the steel envelope will also be studied.

Figure 2: Artist's impression of OPERA supercontainer for heat-generating HLW



The chosen dimensions (smaller containers, shorter disposal drifts) reflect the amounts and characteristics of the waste in the Netherlands and facilitates the basic requirement of waste retrievability. In 1993 the Dutch government adopted a position paper (VROM, 1993) on the geological disposal of radioactive and other highly toxic wastes. That position paper was presented to parliament, and forms the basis for the further development of the national radioactive waste management policy: any underground disposal facility to be constructed shall be designed in such a way that each single step in the disposal process can be reversed. One of the consequences of that position is that, if deemed necessary for whatever reason, retrieval of the waste must be possible even after closure of the repository.

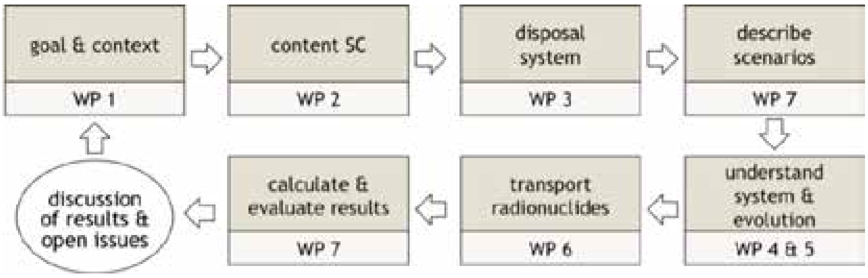
OPERA research topics

The OPERA research programme consists of over 40 complementary tasks with well-defined content and clear interfaces with other tasks, grouped in seven work packages (WP, see also Figure 3), each concentrating on a particular aspect of the safety case (Verhoef and Schröder, 2011):

- *WP1: Goal & Context* – Defines all contextual and logistic boundary conditions for the OPERA safety case. Topics that are treated include waste characteristics, political requirements and societal expectations, and communicating the safety case.
- *WP2: Content Safety Case* – Has the central role of setting up and defining the two post-closure safety cases, for Boom Clay and rock salt. It elaborates the structure and methodology of the OPERA safety cases in more detail, and it outlines the framework of the safety assessment. This paper describes the results obtained so far in the OSCAR project in Work Package 2.
- *WP3: Disposal System* – Evaluates the principal feasibility of a disposal concept in Boom Clay in the Netherlands at 500 m depth. In addition, possible design modifications may be investigated that may reduce uncertainties associated with the safety assessment of the system concept.
- *WP4: Geology and Geohydrology* – The long-term safety of the geologic disposal is evaluated, considering past and possible future evolutions of the geosphere.
- *WP5: Geochemistry and Geomechanics* – Defines the geochemical/geomechanical properties of the undisturbed Boom Clay. In addition, the most relevant degradation and corrosion processes of materials that are part of the EBS or waste fraction, as well as their interaction with Boom Clay are investigated.

- *WP6: Transport Radionuclide* – Addresses the processes controlling the transport of radionuclides from the waste container, through the Boom Clay and surrounding rock formations, into the biosphere.
- *WP7: Scenario Development and Performance Assessment* – Comprises the overall radiological long-term, post-closure safety assessment, and makes use of the input generated within the other OPERA Work Packages. The safety assessment will be performed with the open source reactive-transport modelling framework ORCHESTRA (Meeussen, 2003), which – if necessary – will be adapted to the specific needs of the defined physical systems in the individual scenarios.

Figure 3: Organisation of the research programme in seven work packages



The OPERA safety case

The structure of the initial long-term, post-closure safety case for a disposal facility for radioactive waste in Boom Clay in the Netherlands is being developed in the OSCAR project. The OPERA safety case (OSC) is based on the evaluation of the scope, structure and argumentation of existing international safety cases, safety reports and license applications and identifies the best-suited elements for the Dutch programme.

Although the OPERA research programme is primarily focused on the disposal concept in Boom Clay, part of the management strategy in the Netherlands is also aimed at developing and maintaining the knowledge of radioactive waste disposal in rock salt as has been developed in previous Dutch national research programmes. The OSC will serve as a basis for the further development of the subsequent stages of the Dutch radioactive waste disposal programme (cf. Figure 4, modified from IAEA, 2012).

The OSCAR project addresses the following topics for a potential repository in Boom Clay:

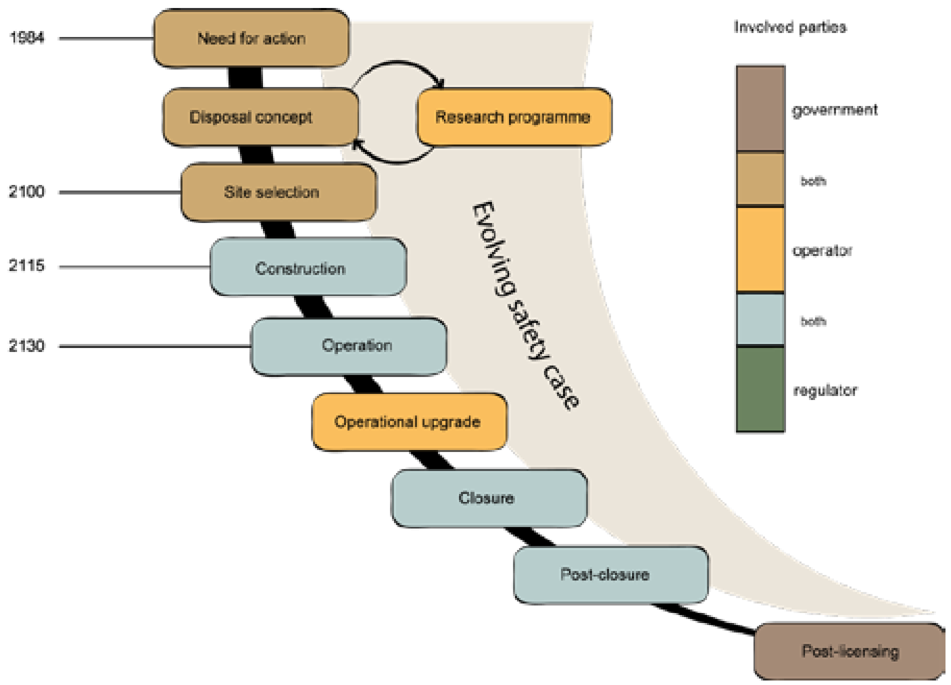
- *Evaluation of the state-of-the art on safety case methodologies* (complete). This task comprises comparison and evaluation of the structures and methodologies of relevant safety cases developed in other countries as well as the guidelines provided by the NEA and IAEA. The aim of the evaluation is to collect information, to get acquainted with the safety case concept, and to provide a basis for the subsequent development of the OPERA safety case.
- *Proposal on structuring the OPERA safety case* (in progress). The OSCAR consortium develops a proposal for the structure of the OPERA safety case. The basis of the safety case structure is found in the information and ideas collected in the previous task and draws on the experience gained from the IAEA-hosted project PRISM[5] (Nys, 2012).

5. Practical Implementation of Safety Assessment Methodologies in a Context of Safety Case (PRISM).

- *Organise and structure the OPERA research efforts using safety statements* (ongoing). The concept of safety statements was developed by ONDRAF/NIRAS. Safety statements describe the claims about the safety of a geological disposal facility and relate these claims to supporting evidence and sub-claims. Safety statements are also valuable in the sense that safety functions of the disposal system are explicitly incorporated. In co-operation with ONDRAF/NIRAS, the concept of safety statements is developed and applied in the Dutch context.
- *Safety assessment methodology* (complete). This defines the overall methodology and strategic framework for the OPERA safety assessments. It provides a high-level description of the assessment strategy and describes approaches for quantifying the behaviour of a repository, including the treatment of uncertainty. As part of this work, the most relevant existing safety assessment methodologies are compared and evaluated. This exercise guarantees that all safety assessment aspects deemed relevant nowadays are incorporated into the OPERA safety assessment activities.
- *Features, events and processes (FEP analysis*) (midway). This task covers a FEP analysis for the OPERA disposal concept (Verhoef, 2011) to identify and evaluate a set of repository evolution scenarios concerning consistency and applicability within the Dutch context. Again, advantage is taken from existing waste programmes in other countries, as well as the NEA, the IAEA, and the European Atomic Energy Community's Framework Programmes.

The following sections elaborate on most of these OSCAR topics in more detail.

Figure 4. Evolution of a safety case for geological disposal

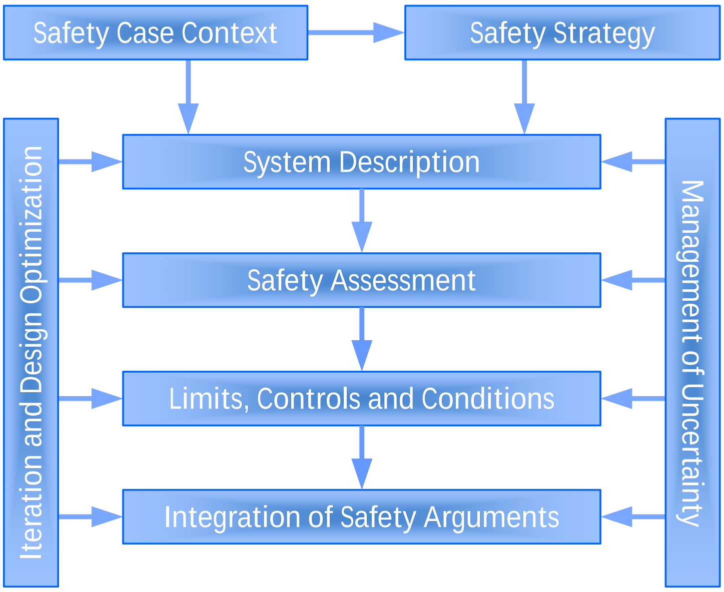


Evaluation of the state-of-the art on safety case methodologies

In this task the structures and methodologies of safety cases developed in Belgium, Finland, France, Germany, Sweden, Switzerland and the United States were compared and evaluated. Moreover, an evaluation of a Dutch safety case for the underground storage of CO_2 was included, to compare similarities from a different perspective and to learn lessons from that case that could be of value for the OPERA safety case. The objective of this task was to provide a basis for the subsequent development of the OPERA safety case concerning its structure and basis of guiding the research efforts.

The review of these safety cases revealed that there is no universal format or plan for achieving and documenting a safety case. However, there is international consensus on the main safety case elements. The NEA and IAEA guidelines both reflect an international consensus on what should be the main elements in safety case. As the design of OPERA was based on the elements of a safety case from NEA guidelines (2004), in OSCAR this was compared to IAEA guidelines (IAEA, 2012). Figure 5 gives a schematic overview of safety case development according to the IAEA guidelines.

Figure 5: Components of a safety case according to the IAEA (2012)



Note that the arrows to the left and right refer to what IAEA calls "interacting processes" – not safety case structure.

Proposal on structuring the OPERA safety case

For the comparison with the IAEA guidelines, the outcomes of the IAEA-hosted project PRISM (Nys, 2012) were also used in the OSCAR project. PRISM paid special attention to the confluence of the main decisions taken in a radioactive waste disposal programme and the safety case arguments involved in each decision. It details a management system and process for interaction with the regulatory body and interested parties. As such the IAEA/PRISM methodology is valuable in cataloguing the relevant aspects of the safety case for a disposal facility for radioactive waste. To check for completeness, all of the tasks described in the OPERA research plan (Verhoef and Schröder, 2011) were projected on the components.

Application of the IAEA/PRISM methodology shows that the OPERA tasks can relatively easily be projected on IAEA components. A task in OPERA related to the component "Limits, Controls and Conditions" (cf. Figure 5) appears to be missing. However, in the present early stage of the Dutch geological disposal programme no clear regulatory guidelines on "Limits, Controls and Conditions" have yet been developed. Consequently, the investigations on their development are covered in the structure component "Safety Case Context".

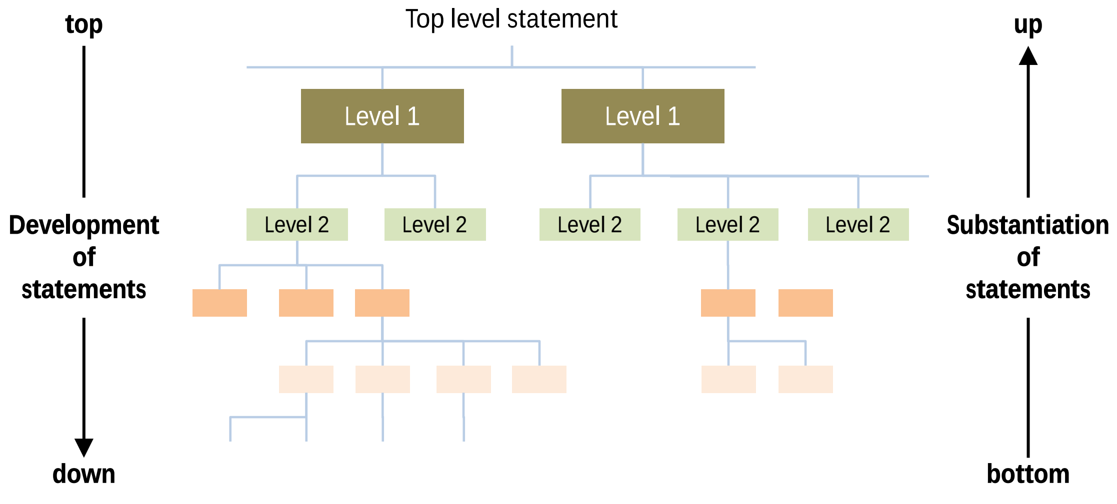
Based on the evaluation of the diverse safety cases as well as the IAEA (PRISM) guidelines and NEA documentation a proposal is being developed in OSCAR, detailing a structured framework for documenting and presenting all of the safety relevant information from the over 40 tasks in a consolidated manner.

Organise and structure the OPERA research efforts using safety statements

The safety of deep geological disposal depends on a variety of aspects, which all need to be addressed adequately and sufficiently in a safety case for the geological disposal of radioactive waste. The expectations of diverse stakeholders, e.g. politicians, the public etc., also need to be taken into account. This implies that managing all safety-related information and knowledge in a clear, understandable and well-structured manner is crucial. One manner to accomplish this is the formulation and assessment of so-called safety statements.

The concept of safety (and feasibility) statements was developed by ONDRAF/NIRAS as a means to describe claims about safety aspects of a geological disposal facility, and to underpin these claims with supporting evidence and sub-claims. The safety statements are structured in a top-down manner, starting with the most general (high-level) statements and progressing to increasingly specific (lower-level) statements (Figure 6). The top-level statements define the main objective of the safety assessment of the safety case at hand, namely that the safety concept and the design of the proposed disposal system show sufficient promise to proceed to the next programme stage. The substantiation of the statements, with multiple lines of evidence and their associated uncertainties generated from the RD&D programme, is performed bottom-up. The need to obtain arguments to substantiate the lowest-level statements and to address any open issues guides the Belgian RD&D programme, ensuring that the focus is maintained on the upper claims.

Figure 6: The top-down development of the structured set of safety statements and the bottom-up assessment of the level of support for these statements



In Belgium, four branches of safety statements are presently being considered (Smith, 2009) supporting the statements:

- The system is known.
- The safety functions that have been defined are relied upon.
- The performance of the disposal system meets the requirements.
- Remaining/residual uncertainties.

Within OSCAR it is investigated to what extent the Belgian safety statements (Smith, 2009) can be used directly in OPERA, to what extent they need to be modified and how they can be used to effectively organise the outcomes of the different research tasks. Important for this task is that the safety statements and the related substantiation must reflect the stage of the programme and aim of the safety case and – in particular the lower level – will develop over time.

OSCAR has started the task by comparing the Belgian statements with the OPERA research tasks defined in the OPERA research plan (Verhoef and Schröder, 2011). A further objective is to formulate an initial, generic set of safety statements relevant for OPERA. A meeting with representatives of all of the projects is planned to discuss the set of safety statements and the linkages to the different tasks.

Safety assessment methodology

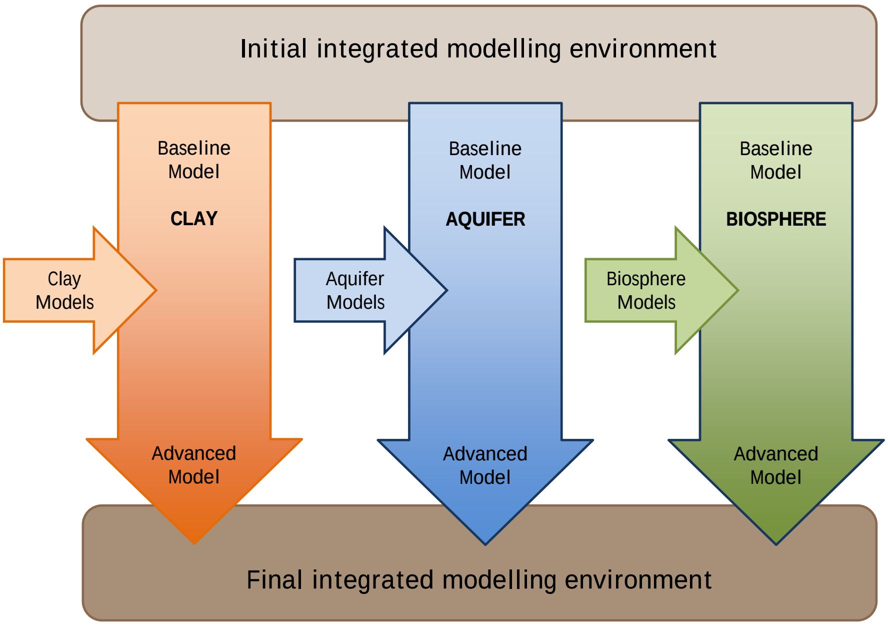
Safety assessment methodologies of the IAEA, NEA, FP6 project PAMINA, as well as the national approaches of Belgium, Finland, Germany, the Netherlands and the United States have been used as a starting point for setting out the OPERA safety assessment methodology. The Dutch safety assessment strategy for CO_2 storage (Barendrecht) has also been evaluated. It was concluded that, although details of the approaches of the respective safety assessments may differ [e.g. concerning the development of scenarios, the application of FEP, the approach of probabilistic methods (if applicable), the use of safety and performance indicators], the overarching key steps or components are similar and in compliance with generally accepted methodologies as reported by PAMINA (Galson, 2011), the NEA (OECD/NEA, 2012) and the IAEA (2004).

Application of the OPERA safety assessment methodology comprises the following elements:

- evaluation of the source term (radioactivity, matrix compositions);
- identification of FEP – features, events and processes;
- scenario development;
- scenario representation;
- development of a PA model for radionuclide migration in Boom Clay;
- definition of safety and performance indicators calculation methodology;
- determination of the methods to be used for the uncertainty analysis;
- development of the performance assessment model for radionuclide migration in the rock formations surrounding the host rock;
- development of the performance assessment model for radionuclide migration and uptake in the biosphere;
- development of an integrated modelling environment for safety assessment;
- parameterisation of performance assessment models;
- safety assessment calculations;
- interpretation and evaluation of the results;
- quality assurance.

The procedure that is being applied in OPERA to develop the integrated post-closure performance assessment (PA) model is schematically depicted in Figure 7. The development starts with formulating an initial integrated model using relatively simple PA model presentations of the three main compartments clay (including the waste and

Figure 7: Procedure to develop the integrated performance assessment model



the EBS), the aquifer system and the biosphere. These baseline models are based on existing safety assessment efforts in the (previous) Dutch, German and Belgian research programmes. The integrated model takes into account the prevailing boundary conditions of the disposal concept, as well as the requirements and safety strategy as currently implemented in the Netherlands.

As the OPERA research programme progresses and results from detailed studies become available, the PA compartment models gradually advance to the final integrated modelling environment. The advanced PA compartment models will be based on and/or calibrated by the detailed models developed within the OPERA programme.

Concluding remarks

The structure of the initial long-term, post-closure safety case for a disposal facility for radioactive waste in Boom Clay in the Netherlands is being developed in the OSCAR project. Hereto a selection of relevant national and international efforts concerning the set-up of a safety case for geological disposal of radioactive waste (safety case structure, safety assessment methodology, FEP database) has been reviewed considering the objectives and outlines of the OPERA programme described in the OPERA research plan (Verhoef and Schröder, 2011). Not surprisingly, it turned out that the guidelines and databases of the IAEA and NEA developed by the international community pretty well covered all aspects of nationally developed safety cases.

Although in OPERA only “initial and conditional” safety cases (for disposal in low permeable clay and rock salt) will be developed, the programme objective is detailing a first roadmap for the long-term research on geological disposal of radioactive waste in the Netherlands. The safety case being developed will serve as a basis for the further development of the subsequent stages of the Dutch radioactive waste disposal programme. The focus of OSCAR is, therefore, to develop and propose a “future proof” structure for the safety case, drawing on the NEA and IAEA/PRISM methodologies. The OPERA safety case structure being developed will encompass all relevant aspects, or components, of a modern safety case and will link the different components in a practical and transparent way. It will assist in steering the flow of information generated within the different OPERA and as such provide a structured framework for documenting and presenting all of the safety relevant information from the over 40 tasks in a consolidated manner. Central to the safety case lies the methodology of the probabilistic safety assessment. The proposed safety assessment structure is derived from the recent national and international efforts in order to include all relevant aspects deemed necessary and facilitates integration of the results of the different tasks in the safety assessment model environment.

As OPERA builds on the ONDRAF/NIRAS disposal concept and research, focuses on the same geological formation on Boom Clay, collaboration is clearly advantageous for both parties. In the Belgian RD&D programme the concept of safety statements is being developed to manage information and focus the research to be relevant to support the safety case. To strengthen the collaboration between both RD&D programmes, this concept will be applied and further developed in OPERA as well.

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Czech safety concept: 2013 state of the art

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Introduction

The Czech Republic operates four WWER 440 reactors (Dukovany) and two WWER 1000 reactors (Temelín). The four 440 MW Dukovany units were installed and began operation during the period 1985-1988. The two WWER 1000 reactors at Temelín started operation in 2002 and 2003. Currently, more than 8 000 SF assemblies from WWER 440 reactors and 900 spent assemblies from WWER 1000 reactors spent fuel assemblies are stored in dry storage facilities located in the area of both NPP in approved casks or in pools at reactor sites. More than 4 000 assemblies are expected to be spent by 2025 at Dukovany reactors and 4 600 assemblies by 2042 at Temelín reactors. The multi-billion Euro contract to build two new nuclear reactors at the current site of Temelín with the option for an additional one in Dukovany has recently been launched in the Czech Republic. It is expected that more than 8 000 fuel assemblies would be spent in the three new nuclear reactors in the Czech Republic during their 60 years of electricity production. The basic reference plan is to directly dispose of all of the spent fuel assemblies in a deep geological repository (DGR), starting operation not earlier than in 2065.

The DGR is planned to be located in granite host rock, because no other type of host rock in sufficient volume is available in the Czech Republic. Currently seven candidate sites for DGR suitable for geological disposal of SF assemblies have been selected, but due to negative community attitudes at the notion of have a repository in their backyard, they are still awaiting a detailed geological survey. According to proposed reference designs, SF assemblies should be in steel-based canisters emplaced in vertical or horizontal boreholes in granite host rock at approximately 500 m under the surface and surrounded by compacted bentonite.

The Czech safety concept is based on the KBS-3 concept developed in Sweden. The Swedish concept is primarily based on almost thermodynamic stability of copper overpack in reducing conditions of granite host rock 500 m under the surface and on almost impermeable compacted bentonite surrounding canisters that protect canisters against corrosion. It is supposed that the corrosion rate of copper under repository conditions is so slow that no canister would fail in 10^6 years (SKB, 2010). Only under some not very probable conditions connected with erosion of the surrounding bentonite canisters, it was estimated that corrosion could lead to the penetration of several copper canisters within the safety assessment of 10^6 years. Disadvantages of copper canisters are: difficult welding of lids requiring advanced technologies both for welding and welding control and a higher price of copper canisters in comparison to canisters based on steel. This was the reason why the Czech safety concept proposed cheaper steel-based canisters instead of copper-based canisters. It is well known that the steel is not thermodynamically stable under reducing conditions, but it is known that under anaerobic conditions its corrosion rate can be very slow if microbial corrosion can be excluded (the same condition also applies for copper canisters). This is possible due to the compacted bentonite preventing growth of microbes. Contrary to Sweden the Czech Republic lies in geologically more favourable conditions with respect to canister corrosion because usually the groundwater contains lower concentrations of the chlorides that are the most aggressive agent for corroding stainless steels than groundwater at sites near the sea. The reference canister design in the Czech DGR concept should be composed of two layers with one outer layer from carbon steel corroding very slowly under anaerobic conditions and a second inner layer from stainless steel corroding with almost negligible general corrosion rate under anaerobic conditions and low tendency to local corrosion under anaerobic conditions. The major challenge in the Czech safety concept, otherwise very similar to the Swedish one, is therefore to prove that steel-based canisters can meet all the requirements with sufficient margin. A comprehensive research and development programme for steel-based canisters is under preparation.

In this contribution the results of the preliminary safety case of the Czech safety concept with steel canisters are summarised.

Methodology and data used for safety case

Methodology

After analysis of OECD/NEA FEP database (2000), the following six scenarios of possible repository evolution under geological conditions of the Czech Republic were derived for more detailed calculations:

1) Central scenario involving all FEP which can likely occur in the repository system (Central).
2) Scenario initiated by earthquake leading to immediate failure of a various number of canisters in one year (Earthquake 1).
3) Scenario initiated by earthquake leading to immediate failure of a various number of canisters in one year and simultaneously to advective flow in bentonite surrounding these canisters (Earthquake 2).
4) Scenario initiated by change of climate leading to reducing by half the outflow of water from the repository to the surface (Climate).
5) Denudation and erosion scenario leading to reducing by half the length of the pathway from the repository to the surface (Erosion).
6) Human intrusion scenario leading to the failure of one canister due to geological survey 300 years after repository closure (Intrusion).

Two approaches have been used for calculations: screening calculations using a MS-Excel spreadsheet and calculations with the GoldSim transport code. Details of these approaches are included in reports of the project “Update of DGR Reference Design in a Hypothetical Site” completed in 2011 (Vokál, 2009, 2010). The screening calculations conducted only for the central scenario were used for a comparison of the following disposal solutions:

1) direct disposal of SF assemblies in steel-based canisters in a vertical arrangement (A-Fe-V);
2) direct disposal of SF assemblies in steel-based canisters in a horizontal arrangement (A-Fe-H);
3) direct disposal of SF assemblies in copper-based canisters in a vertical arrangement (A-Cu-V);
4) direct disposal of SF assemblies in copper-based canisters in a horizontal arrangement (A-Cu-H);

5) disposal of SF assemblies from current reactors directly in steel canisters in a vertical arrangement and disposal of remains of SF assemblies from new reactors after reprocessing in vitrified form and in the form of MOX fuel (B-Fe-V);

6) disposal of SF assemblies from current reactors directly in steel canisters in a horizontal arrangement and disposal of remains of SF assemblies from new reactors after reprocessing in vitrified form and in the form of MOX fuel (B-Fe-H).

The screening calculations enable to estimate maximal effective doses only, but their great advantage is that you can easily follow all the assumptions and simplifications accepted in calculations.

In the second approach, used for calculations of all the scenarios give above, GoldSim contaminant transport computer code together with hydrogeological code FLOW 123D (Vokál, 2009) was used for calculations. The advantage of this method is that the evolution of effective doses can be followed over time, radionuclide ingrowth in radionuclide chains can be incorporated, and the modelling of some processes, such as diffusion of radionuclides in granite matrix, can be incorporated in more sophisticated way.

Data summarisation

Inventory of radionuclides in SF matrix, structure materials and HLW decommissioning waste was taken from previous inventory calculations.[1] Inventory of instant release fraction was taken from Johnson, *et al.* (2005). The matrix degradation rate 1×10^{-8} yr^{-1} was taken from SKB (2006a) and structure materials and HLW waste degradation rate was selected to be 1×10^{-5} yr^{-1} Vokál (2009).

Data for description of corrosion of steel canisters were estimated on the basis of experiments and analyses conducted in ÚJV Řež, a.s. (Dobrev, 2009; Vokál, 2009). In the reference calculation case, it was expected that canisters would fail to agree with Weibull distribution with a mean lifetime of 110 000 years, a minimum lifetime of 50 000 years and a slope of Weibull distribution 1.3. Copper overpack was represented in screening calculations by a mean lifetime of 5 million years, a minimum lifetime of 0.6 million years, and a Weibull slope parameter 1.3.

Migration parameters for radionuclides (solubility, sorption, diffusion coefficients) in all of the repository components (bentonite, granite, cement) were taken mostly from reports (SKB, 2006a; Bradbury, 2003; Duro, 1997).

For screening calculations the geosphere was represented by three zones:

1) Impermeable zone with average water flux 0.41 m^3/yr and time of flow 4 600 years.

2) Zone with flux of 756 m^3/yr and time of flow 130 years.

3) Zone with flux of 3 000 m^3/yr and time of flow 85 years.

For a description of a hypothetical site in GoldSim calculations available data from geological survey of some granite sites in the Czech Republic (Melechov, Potkůčy and Příbram sites) were used. The hydrogeological simulation code Flow123D, developed at the Technical University of Liberec (Maryška, 2008), was used for hydrogeological description of a hypothetical site. It combines two basic approaches for hydrogeological modelling in a single model, where the rock block represented by a porous medium contains deterministic fractures affecting hydraulically permeable fractures or extensive tectonic zones. On the basis of calculations of transport of non-sorbing contaminants from the repository to the surface, four of the most probable pathways leading through 2-D or 3-D

1. Data provided by ČEZ, a.s., and Burien (1997).

elements from the repository to the surface were selected (Královcová, 2009). The total length, time of flow and total inflow and outflow of water in the selected pathways are showed in the Table 1.

Table 1: Average parameters for selected pathways in geosphere

	Units	G1	G2	G3	G4
Overall length	m	1 803	1 874	862	863
Flow time	yr	7 514	9 554	3 129	5 480
Total geosphere inflow	$m^3.y^{-1}$	52	11	24.4	9.6
Total geosphere outflow	$m^3.y^{-1}$	1 616 914	163 649	23 001	7 901

Similar to screening calculations, it was very conservatively supposed that all contaminants released from the whole repository go only along one pathway. For reference calculations, the pathway (G4) with lowest outflow (and the lowest dilution) was selected. The data for migration of radionuclides into granite matrix were taken from IAEA (2010).

A stylised resident scenario with an agricultural farm built on the area of radionuclide discharge was used for biosphere calculations. Most of the data for biosphere calculations were taken from Vieno (1992). Details are given in Vokál (2009).

Results and discussion

Maximal annual effective doses of various disposal solutions calculated in screening calculations are given in Table 2.

Table 2: Maximal effective doses of various disposing options

Notation of solution	Short description of solution	Maximal effective dose (mSv/yr)
A-Fe-V	SF assemblies in steel canisters in vertical arrangement	0.140
A-Fe-H	SF assemblies in steel canisters in horizontal arrangement	0.021
A-Cu-V	SF assemblies in copper canisters in vertical arrangement	0.027
A-Cu-H	SF assemblies in copper canisters in horizontal arrangement	0.0034
B-Fe-V	Part of SF assemblies is reprocessed and dispose of in vitrified form, otherwise the same as A-Fe-V	0.110
B-Fe-V	Part of SF assemblies is reprocessed and dispose of in vitrified form, otherwise the same as A-Fe-H	0.016

The lowest effective doses are obtained with copper canisters in a horizontal arrangement, but the effective doses are not much lower in solutions with copper canisters than in solutions with steel canisters and are greater than the doses derived in safety assessments performed by SKB (2006b). The reason is that the maximal annual doses for the solutions with copper canisters are reached much later than 1 million years, which is usually the outer limit of safety assessment calculations in SKB. The doses are always lower in horizontal arrangement. This is caused by the greater thickness of the bentonite layer surrounding the canisters in a horizontal arrangement than in a vertical arrangement. The greater the bentonite thickness, the slower the migration of mobile radionuclides (^{129}I, ^{36}Cl) into the host rock.

In Table 3 the critical radionuclides together with maximal annual effective doses from screening and GoldSim calculations are compared for the horizontal arrangement with steel canisters (A-Fe-H) (the thickness of the bentonite layer was 0.7 m in GoldSim calculations instead of 1 m in screening calculations). For GoldSim calculations the time of reaching annual maximal doses is also given. Only radionuclides with annual effective dose higher than 1×10^{-5} mSv/yr are shown. The calculations were conducted up to 1 million years.

Table 3: Critical radionuclides from screening and GoldSim calculations

Screening calculation		GoldSim calculation		
Radionuclide	Effective dose (mSv/yr)	Radionuclide	Effective dose (mSv/yr)	Time of peak (yr)
^{129}I	2×10^{-2}	^{129}I	8×10^{-2}	8×10^{4}
^{36}Cl	1×10^{-3}	^{36}Cl	2×10^{-3}	8×10^{4}
^{135}Cs	2×10^{-4}	^{79}Se	3×10^{-3}	7×10^{5}
^{126}Sn	7×10^{-5}	^{135}Cs	4×10^{-5}	1×10^{6}
^{79}Se	2×10^{-5}	^{41}Ca	3×10^{-4}	4×10^{4}
		^{14}C	3×10^{-4}	4×10^{4}

It can be seen that despite different calculation approaches and also small differences in data (e.g. thickness of bentonite, different data for the description of pathway in geosphere), the results of calculations are very similar. Only five or six radionuclides are important for the long-term safety of a geological repository. The predominant effect of ^{129}I, with a half-life of 1.57×10^{7} years, on annual effective doses can also be seen. The contribution of ^{41}Ca and ^{14}C from decommissioning waste (calculated only using GoldSim) is very small, but not negligible.

In Table 4, the results of calculations for various scenarios of the repository evolution described above are given.

Table 4: Results of GoldSim calculations for difference scenarios of repository evolution

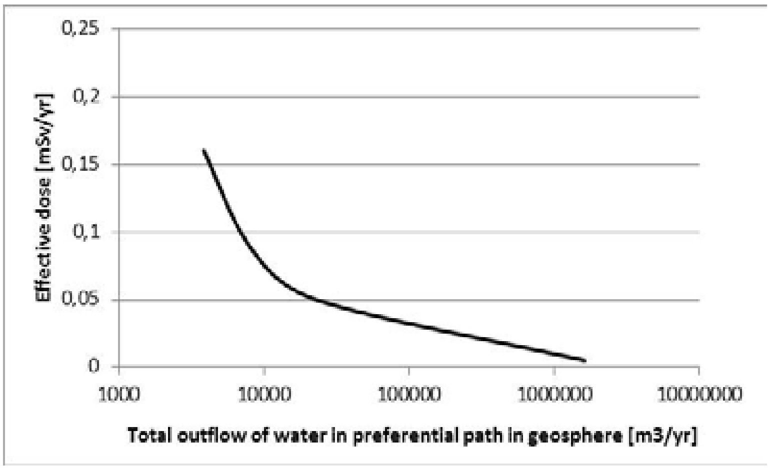
Notation	Short scenario description	Effective dose (mSv/yr)
Central	Normal evolution scenario with high probability FEP, but not considered any low probability event or process	0.08
Earthquake1	Earthquake leading to the immediate failure of 1 000 canisters in one year at time corresponding to mean lifetime of canisters	0.18
Earthquake2	Earthquake leading to the immediate failure of 1 000 canisters and failure of bentonite buffer causing advective flow in one year at time corresponding to mean lifetime of canisters	0.48
Climate change	The change of dilution effect (to half) due to climate change	0.16
Erosion	The change of length of pathway (to half) due to erosion	0.08
Intrusion	The failure of one canister and bentonite due to geological survey 300 years after repository closure	0.08

Only under a very low probability scenario (Earthquake2) in which 1 000 canisters would fail in one year and simultaneously an advective pathway through bentonite was formed, did the annual effective dose exceed the regulatory limit of 0.25 mSv/yr.

The central evolution scenario on which other scenarios are based do not represent the normal evolution scenario because, mainly due to limited knowledge, very conservative assumptions are accepted. For example, the assumption that all radionuclides will go only along one pathway or that radionuclides from all waste packages will find their pathways to the surface does not seem very probable, but cannot be excluded by current knowledge. Also in the description of corrosion of steel canisters no credit was taken for dry sites in which corrosion would be limited by water shortages or by iron transport from canisters (Vokál, 2005).

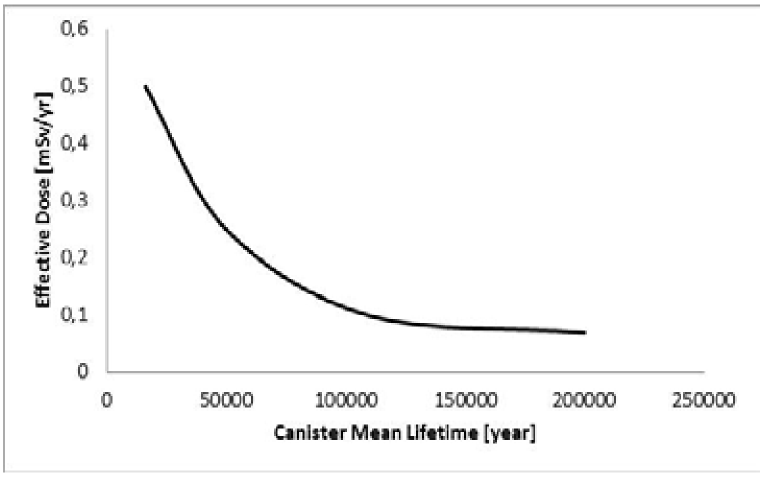
The greatest uncertainties are related to canister behaviour and geosphere properties. The most important parameter concerning geosphere seems to be the outflow of water in the pathway from the repository to the surface leading to mixing of contaminated water with uncontaminated water on the way to the surface. The effect of change of the outflow of water in pathways (affecting dilution) is shown in Figure 1. The greater the outflow of water, the greater the dilution and the lower annual effective doses.

Figure 1: The effect of outflow of water along the pathway from the repository to the surface



In most calculations, canisters are represented with a mean lifetime of 110 000 years, a minimum lifetime of 50 000 years and a 1.5 slope of Weibull distribution. These properties are not, however, substantiated by a sufficient amount of data. The most problematic is primarily determination of distribution of canister failure. If immediate failure of all the canisters is considered in calculations, the annual effective dose is only moved to the right in the timeline, because annual effective dose is governed by ^{129}I with a half-life of 1.57×10^7 years. The corrosion of metals is invariably connected with the spread of results. Even if a small number of samples is immersed under the same conditions in water, different results are always obtained for individual samples. The same must apply for canisters in a repository. One must, however, be very prudent in the selection of canister failure distribution, because using too wide a distribution could lead to risk dilution. For illustration purposes the dependence of effective doses on the mean lifetime of canisters with the same ratio of standard deviation to the mean lifetime of canisters (5.8) in Gauss distribution is shown in Figure 2. Under these conditions the effective dose decreases rapidly with the increase of mean canister lifetime until values around 100 000 years due to the increase of the spread of canister failures.

Figure 2: The dependence of effective dose on mean lifetime of canisters



Conclusions

The steel canister Czech safety concept requires a long-term research programme focusing on the determination of canister properties under repository conditions. The advantage of this safety concept, contrary to that of copper canisters, is that the disposal of spent fuel assemblies in steel canisters can be more economical. Since the start of repository operation in the Czech Republic is planned to be in 2065, there seems to be sufficient time both to implement this research programme and to return to the more expensive copper canister solution in case of failure to prove that disposing of SF assemblies in steel canisters in a granite host rock is safe.

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Safety cases and siting processes

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Central to any process for building a deep-mined geologic repository for high-activity radioactive waste is the development of a safety case.[1] To date, such cases, in various forms have been elaborated for a variety of concepts for geologic disposal, including in salt, clay, argillite, crystalline rock (granite and gneiss) and volcanic tuff formations. In addition to the technical effort required to develop a safety case, increasingly nations have come to believe that it is also critical to obtain the consent of the region or community[2] where the facility might be located. The purpose of this paper is to explore issues associated with just one aspect of consent-based siting: How can such a process be designed so that *willingness* to accept a site for a repository continues to be meaningful even as new technical knowledge and insights emerge during site characterisation? In short, what is the meaning of "informed consent" in the context of repository development?

The timing of consent

In countries that have selected a site for a deep-mined geologic repository, the timing of consent varies. In France, the village of Bure volunteered to host an underground research laboratory with the understanding that a repository might subsequently be developed nearby (JORF, 1991). In Finland, consent is given when a "decision-in-principle" is taken by government (Finlex, 1987, 1988). That action is based solely on preliminary surface-based investigations. In Sweden, the municipalities must agree to government granting a license to construct the repository (SFS, 1984). Again, permission is given based only on data collected from the surface. The United States is the one nation where consent is sought after site characterisation investigations, including studies at the emplacement horizon, have been completed (USC, 1983).[3] In nations such as Canada and the United Kingdom

1. "An integration of argument and evidence that describe, quantify, and substantiate the safety and the level of confidence in the safety of a geological disposal facility. 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, and also on the research programme and management strategy by which uncertainties and open questions are handled." (OECD/NEA, 2009, p. 15) Safety cases can include probabilistic and non-probabilistic performance assessment but only as one element within a larger set of arguments.
2. Throughout this paper, we use the generic terms "community" and "locality" to refer to some unit of government below the national level. Precisely what type of jurisdiction these terms refer to will vary from country to country.
3. It is important to note that although the Nuclear Waste Policy Act seeks to secure a state's consent, consent was not given in the case of the proposed Yucca Mountain repository. Instead, the US Congress overrode the state of Nevada's veto of the site recommendation. The veto power that a state possesses is therefore much weaker than the veto power a community possesses in either Finland or Sweden.

(except Scotland) that have established siting processes but have not selected a site, communities initially express interest, but they maintain the right to withdraw up to some predetermined point, typically when large investments need to be made to construct underground workings.

The fundamental tension

As site characterisation progresses from literature reviews to surface-based studies to underground investigations at depth, knowledge accumulates. For a community concerned about the possibility of surprise, postponing final consent to as late a stage as possible makes sense. The community can then evaluate how the implementer has resolved outstanding technical questions. At the same time, a community must be sensitive to the possibility that, as the characterisation proceeds, momentum for the project may build to the point where it is difficult to reverse course and withdraw consent, particularly in the face of accumulating sunk costs. In contrast, the implementer wants consent to be exercised early in the process. The implementer worries about investing significant sums to get underground only to discover that the potential host has decided to object even before the regulatory authority has passed judgment.

Safety cases may not be static

By their very nature, safety cases resist fundamental changes. But such shifts are not unknown. The Swedish disposal concept and associated safety case, the KBS-3 method, dates from the mid-1970s. Under prevailing Eh/pH conditions and given the chemical composition of the groundwater found in the granitic formations where the repository would be located, the elemental copper waste canisters are not expected to corrode. Further protection would be gained by placing bentonite clay, which retards the movement of water, around the canisters. Some recent experiments have challenged the corrosion resistance of the canisters and the effectiveness of the bentonite barriers. Although these studies are by no means definitive, they could lead to modifications in the KBS-3 disposal concept (SNCNW, 2009).

The argument in favour of the proposed Yucca Mountain repository, for instance, originally rested upon the belief that water passed slowly from the surface through various layers of volcanic tuff until it reached the emplacement horizon. Once the Exploratory Studies Facility was constructed and access to the horizon gained, experiments seemed to suggest that water moved more rapidly than expected through rock fractures. This finding led directly to a reassessment of a key parameter of the safety case, percolation flux, and prompted a decision to construct an elaborate engineered barrier system, composed of corrosion-resistant waste packages and drip shields (US NWTRB, 2003).

Safety cases differ in terms of their evidentiary support and robustness. Many of these safety cases have been subjected to international peer reviews (OECD/NEA, 2002, 2003, 2004, 2006, 2012). Some of them, such as those that envision a deep-mined geologic repository sited in either clay or salt formations, are closely tied to measurable physical properties of the host rock (Andra, 2005; ONDRAF/NIRAS, 2001; NAGRA, 2002). In contrast, the safety case for a repository located at Yucca Mountain rested upon claims made about complex interactions between the geologic and engineered barriers that would arise under above-boiling operating conditions (US DOE, 2008).

Site characterisation and the potential for surprise

Site characterisation progresses in three stages, each with the potential for major discoveries or surprises. In the first stage, different types of geology may be evaluated according to their generic or assumed properties, such as the sealing characteristics of plastically deforming salt or the slow movement of ground water through indurated clays. In some countries, only one type of geology may be available, and thus surveys of different locations focus on disqualifying characteristics, such as fracture zones that facilitate rapid transport of water. During this initial stage, most of the geologic data will come from detailed surface mapping followed by a variety of remote-sensing (e.g. mapping fractures using sensors at different wavelengths) or geophysical surveys (e.g. magnetic and radiometric surveys from the air or land-based seismic profiling). Based on the results from this first stage, blocks of potentially suitable geology can be identified on a scale of some kilometres.

The second stage heralds a more detailed examination by drilling that provides actual rock core samples at depth and access via down-hole techniques in order to obtain more detailed information on the characteristics of the rock. At this stage, the three-dimensional properties of the site begin to emerge. Pump tests at wells can be used to assess the hydrologic properties of the site, and samples of water can be dated to determine the degree of isolation from the near-surface biosphere. The exact placement of the repository horizon may be adjusted during the second stage in order to avoid faults, fracture zones, or less desirable rock types. The final, or third, stage of investigation requires underground workings that may be the first step in the construction of the repository. It is only at this last stage that a clear picture of the geology of the site at the repository horizon can be obtained.

At each of the three stages, there can be substantial deviations from the originally envisioned safety case. As one revelation piles on the next, one can imagine that a local community may choose to withdraw from the project. However, the issue for the local authorities is to know when a “surprise” becomes an appropriate reason for withdrawing consent. Such decisions are very difficult to make in the context of a total system performance assessment, extending over hundreds of thousands of years, where the role of engineered barriers may be thought to compensate for some lack of performance from the geologic barriers. Any consent-based process has to empower the local community with enough technical expertise that it can arrive at a satisfactory understanding and confidence in the long-term performance of the repository.

Implicit in the formulation of the fundamental tension is the belief that, once surface-based testing has been completed, the likelihood of securing authorisation to construct and then to operate a repository is extremely high. Since none of the four implementers that have selected sites have received regulatory approval, it is difficult to know how valid that belief might be. However, one should note that these three stages of site characterisation are typical of other geoscience activities, such as the exploration for mineral and hydrocarbon deposits. Huge investments in time and money may precede a “dry hole” at the end of an extended exploration campaign. This is, in fact, a typical outcome. Perhaps the lesson is that success is the surprise – and failure must be anticipated and accepted.

Withdrawing consent

In the four countries where a repository siting decision has been made using consent-based arrangements – Finland, France, Sweden and the United States – consent occurs at a single point and, once given, cannot be withdrawn. Siting processes initiated over the last decade in Canada and the United Kingdom incorporate a two-step approach for signalling consent. Communities invite the implementer to undertake preliminary suitability assessments – mostly literature reviews – to determine whether very general selection guidelines can be satisfied. Based on more extensive surface-based testing, communities decide whether to continue their participation in the siting process.

Under the Adaptive Staged Management programme in Canada, for example, communities propose terms and conditions on which they would have the project proceed (NWMO, 2005). They then negotiate with the implementer to produce a formal agreement, from which withdrawal is not permitted. Only after receiving such a binding commitment will the implementer commence characterisation at depth.

The Managing Radioactive Waste Safely (MRWS) programme in the United Kingdom lays out in some detail how a community can exercise its right of withdrawal. As part of the programme's surface-based testing (Stage 5), boreholes would be drilled.[4] It is at this point that the fundamental tension explicitly manifests itself. As the White Paper on MRWS puts it:

> *In order to minimise financial risk and uncertainty,* before *the [implementer] embarks on a borehole survey programme, the circumstances in which a post-borehole right of withdrawal might be exercised should be identified…* [emphasis added]
>
> *The requirement to define these circumstances before a borehole programme is likely to be both challenging and beneficial; challenging because it will involve matters of judgement, and beneficial because the definition will focus discussion, enhance understanding and make criteria for a right of withdrawal decision explicit before extensive work has been undertaken.* (DEFRA, 2008, p. 57)

Over a period of slightly more than three years, the West Cumbria MRWS Partnership, composed of local government authorities and a wide range of non-governmental organisations, explored whether to move forward into Stage 4 (let alone Stage 5) of the programme, desk-based studies (West Cumbria, 2012). Ultimately, although the Allerdale and Copeland Borough Councils voted to proceed, a negative vote by the Cumbria County Council blocked further participation. As with many complex decisions, multiple factors led to this outcome. One of the most important, however, was concern that the right of withdrawal could be compromised in the future (West Cumbria, 2012, pp. 60-62).[5]

Institutional requirements for ensuring that consent remains informed

What, then, does "informed consent" mean in the context of a repository development process that is continually generating new, and potentially surprising, information? The MRWS White Paper correctly identified what is challenging and therefore problematic: How can you anticipate the unexpected? How can commitments be made when surprises remain a possibility? One approach, of course, would be to await the final results of in-depth characterisation.[6] More probable, the fundamental tension will be resolved so that the final opportunity to grant consent (or not) will occur before underground investigations begin. Institutionalising several safeguards might make it less likely that a community withdraws from a siting process pre-emptively or prematurely.

1) *The right of withdrawal should be embedded in law.* If the right is simply a matter of policy, it can be modified or reinterpreted. Any legislation would have to address whether the right is conditional or unconditional.

2) *The details of how the right of withdrawal will be implemented in specific cases should be negotiated between the implementer and the community.* Agreements reached ought to be enforceable. Communities should be provided with resources, both legal and technical, so that they can negotiate as equal partners.

4. It is not until Stage 6 that underground construction begins.
5. Those concerns may have been heightened because the MRWS White Paper also stated: "In the event at some point in the future, voluntarism and partnership does not look likely to work, Government reserves the right to explore other approaches." (West Cumbria, 2012, p. 47)
6. This is the approach taken in the United States. But as the Yucca Mountain experience made quite clear, rejection by a state could very well be overridden by Congress.

3) *An independent body should be the final arbiter when differences in interpreting technical information arise.* Well into the siting process in both Sweden and the United States, critical information surfaced that raised fundamental questions about each nation's safety case. That information was subject to differing interpretations. Although such differences in the first instance should be investigated jointly by the implementer and the community, if they remain unresolved the issue should be addressed by an independent disinterested body. That body could be a technical reviewer, such as the National Council for Nuclear Waste in Sweden or the Committee for National Evaluation in France. In any case, the independent body will have to take steps ahead of time to merit the trust and confidence of the community.

4) *At some point, agreed to in advance, a community's option to withdraw can no longer be exercised.* The regulatory authority will need to closely scrutinise the information subsequently developed by the implementer to ensure that "surprises" will not effectively compromise the performance of a proposed repository. As with the independent technical overseer mentioned above, it will be critical that the authority merit the trust and confidence of the locality.

In an ideal world, the fundamental tension would not manifest itself. The implementer and the community would interpret new information identically and would reach a common position about the direction of a repository project. In the real world, however, the implementer may not acknowledge that critical issues have emerged. But even if it does, the implementer has an incentive to "fix" problems as they arise, even if that requires modification of the safety case or interpreting information in the most favourable way. Such adjustments can be entirely appropriate; in fact, they are to be expected in any staged and adaptive siting process. Whether the changes would lead a community to regret the consent it has given at some earlier stage, however, remains an open question.

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Session 7.4

The Broader Perspective: Safety Cases in Areas Other than Deep Geological Disposal

The safety case in support of the license application of the surface repository of low-level waste in Dessel, Belgium

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Executive summary

The modern concept of the safety case, developed by the OECD/NEA for geological repositories of high- and medium-level waste has been successfully applied by ONDRAF/NIRAS for a surface repository for Category A waste (i.e. low-level waste) in Belgium in the current project phase 2006-2012. This resulted in the submission on 31 January 2013 by ONDRAF/NIRAS of an application for a "construction and operation license" to the safety authorities. The benefits of using the notion of the safety case have been that: *i)* safety has been incorporated in an integrated manner within all assessment basis, design and safety assessment activities; *ii)* the process of development of the license application has gained in clarity and traceability; *iii)* the documentation of the license application contains multiple lines of argumentation for safety rather than argumentation based only on quantitative radiological impact calculations. To offer a comprehensive view on the safety argumentation and its development, it has been found useful to develop the argumentation not only along a safety statements structure but also along the safety report structure.

Introduction

ONDRAF/NIRAS, the Belgian Agency for Radioactive Waste and Enriched Fissile Materials, is responsible for the management of all radioactive waste on Belgian territory. In doing so, ONDRAF/NIRAS is aiming at developing and applying sustainable solutions, which can guarantee the protection of humans and the environment, now and in the future. ONDRAF/NIRAS activities integrate four aspects: science and technique, ecology and safety, economy and finance, and ethics and society.

The development of a concept and a design for a repository and its safety assessment is an iterative process, punctuated by the submission to the government and/or regulatory authorities of safety cases supporting the decision to proceed to the next programme stage. A safety case consists of a set of key documents containing the safety arguments and the key elements of the substantiation of these safety arguments and supporting documents.

A safety case (Safety and Feasibility Case 1) (Capouet, 2014) is currently being prepared by ONDRAF/NIRAS for Category B and C wastes [corresponding to intermediate- and high-level wastes according the classification in the IAEA General Safety Guide GSG-1 (2009)].

The modern concept of the safety case, developed by the OECD/NEA (2004) for geological repositories of high- and medium-level waste, has been successfully applied for a surface repository for Category A waste [low-level waste (IAEA, 2009)] in Belgium in the current project phase 2006-2012. This resulted in the submission on 31 January 2013 by ONDRAF/NIRAS of an application for a “construction and operation license” to the safety authority, the Federal Agency for Nuclear Control (FANC) (ONDRAF/NIRAS, 2013).

Such widening in application of the safety case notion towards facilities other than geological repositories is an example of practical application of an emerging international consensus on a widened scope of use for safety cases reflected for example in the 2011 IAEA Specific Safety Requirements SSR-5 related to the disposal of radioactive waste in general.

This article focuses on the return of experience of the development of the safety case for Category A waste.

Structure of the safety case for Category A waste and reviews before submission of the license application

To accommodate different types of audiences, the safety case for Category A waste is composed of four levels of documents with an increasing breadth and depth of information and scientific and technical detail as the level number increases.

Level 1 is composed of a non-technical synthesis targeted at the general public (ONDRAF/NIRAS, 2013), an overview of the safety argumentation (ONDRAF/NIRAS, 2012r) and a technical summary of the Level 2 (ONDRAF/NIRAS, 2012s).

Level 2 is the safety report and is composed of 17 detailed chapters that serve as a reference for the Safety Report Level 1 (ONDRAF/NIRAS, 2012a-2012q). The safety report contains the safety arguments and the key elements of their substantiation and is written for technical experts.

Levels 3 and 4 supporting documents are the technical reports that have been worked out by and/or on behalf of ONDRAF/NIRAS and bear out the safety arguments described in the Safety Report Level 2. The Level 3 documents describe and substantiate the applied methodologies for development of the scientific basis, the design development and the safety assessments (ASM – assessment methodologies). The Level 4 documents are the application of the methodologies. They describe the scientific and technical basis, the development of the design and its implementation and the development of the safety assessments.

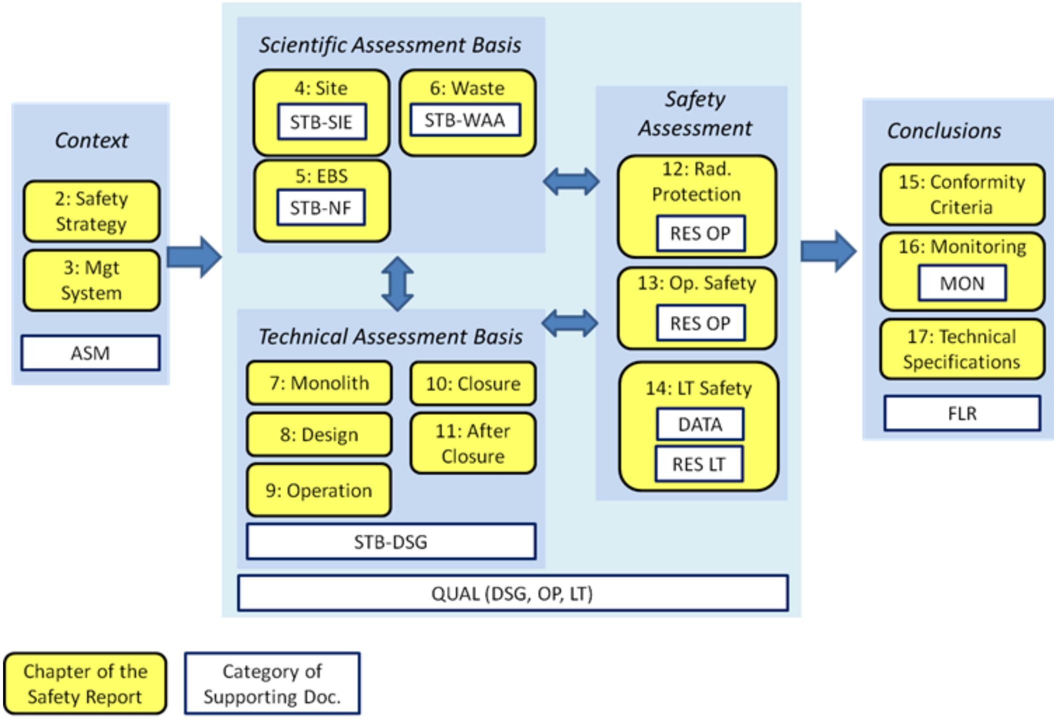
So as to provide traceability of Level 4 documents towards Levels 3, 2 and 1 and towards the safety arguments, a further classification of Level 4 is made into different areas of safety arguments and their substantiation:

- STB: The scientific and technical bases related to: *i)* the site and the environment (STB-SIE); *ii)* the near field, i.e. the multi-layer cover and the cementitious barriers (STB-NF); *iii)* the waste acceptance of category A waste (STB-WAA); *iv)* the design development and description of the components (STB-DSG).
- DATA: The data used for selecting parameter values in the calculations.
- QUAL: The qualification, verification and validation files of models and codes for design development (QUAL-DSG), operational safety assessment (QUAL-OP) and long-term safety assessment (QUAL-LT).
- RES: The results of: *i)* operational safety assessment (RES-OP); *ii)* long-term safety assessment (RES-LT).
- MON: The monitoring programme.
- FLR: The further lines of reasoning including FEP management and uncertainty management.

The target audiences of the Level 3 and Level 4 reports are experts requiring detailed information on the safety arguments and their substantiation and, more generally speaking, experts requiring detailed information on a specific methodological, scientific and technical area referred to in the safety arguments.

Figure 1 illustrates how the organisation of the Level 3 and 4 supporting documents fits within the structure of the safety report.

Figure 1: Organisation of the Level 3 and 4 supporting documents



The safety case has undergone extensive internal and external peer reviews at various levels before its finalisation for the current license application. The various internal and external reviews of the detailed supporting studies and methodologies have been done by a broad involvement of national and international expertise. At a high level, key aspects of the ONDRAF/NIRAS preliminary safety case (version November 2011), have undergone an international peer review organised by the NEA (OECD/NEA, 2012). As a result of discussions during this peer review and as a result of the findings of the peer review, the safety case has been further improved before it was submitted as part of the license application on 31 January 2013. This international peer review has proved to be a fruitful exercise for improvement and clarification of the safety case before it was submitted to regulatory scrutiny by the Belgian nuclear regulator FANC.

Safety strategy

An important preparatory step in the development of a safety case is the establishment of a *safety strategy*, i.e. a high-level integrated approach adopted for achieving safe disposal.

The safety strategy is framed inside an overall iterative management process firmly focused on safety, i.e. the *overall safety approach*. In the overall safety approach, the scientific assessment basis, the design and the safety assessments are all three based upon the safety strategy. This ensures that the developed design and disposal system already take into account all fundamental requirements to ensure safety. With the safety assessments for the developed disposal system, it is checked and confirmed that safety is ensured and the effect of various types of residual uncertainties are estimated as an input for prioritisation and decisions for the next iteration of the disposal programme.

Throughout the development and implementation of the repository, various iterations are performed, ultimately leading to a fully constructed, filled and closed i.e. fully implemented, disposal facility for which there is reasonable assurance of long-term safety. This iterative development and implementation is not only applicable before initial construction, but remains applicable such that before key decisions, e.g. the start of waste emplacement in the repository, the safety case and safety assessments are updated accordingly. Furthermore, a regulatory system of periodical safety revisions is also applied, leading to periodical updates of the safety case.

Through disposal of radioactive waste on the surface, the waste is emplaced in facilities that are directly placed in the biosphere. This implies that the long-term safety after closure of the facility resides on four essential pillars:

1) the properties of the different components of the facility to passively contain and isolate the radioactive waste from humans and the environment;
2) the properties of the disposal site that contribute to this passive containment and isolation (e.g. non aggressive environment);
3) the measures taken to limit the long-lived activity in the waste that can be disposed of, taking into account the containment and isolation performance of the facility;
4) the control and surveillance measures in the repository and its direct surroundings in order to prevent human activities that could perturb the passive containment and isolation provided by the disposal facility.

These pillars rely on the notions of containment and isolation provided by the disposal facility and its constituting systems, structures and components (SSC). The containment and isolation functionalities have been further detailed into *safety functions* that the system of SSC have to fulfil in order to protect humans and the environment, now and in the future.

The safety functions, SSC and different time frames are grouped into the "safety concept" which constitutes a key input towards both design development (establishment of design requirements and conformity criteria) and safety assessment (establishment of scenarios and models based upon the key SSC and safety functions). The safety concept is illustrated in Figure 2.

It is assumed that inadvertent human intrusion by control and surveillance as long as nuclear regulatory control and associated access controls are operational during Phase III (Figure 2). After Phase III, the physical barriers formed by the different components of the facility still provide some means for reducing the likelihood and possible radiological consequences of inadvertent human intrusion.

The safety concept was further detailed in a table describing which systems, structures and components (SSC) have to fulfil which major safety functions during which time frames (see Figure 3). The roles of the various SSC are then categorised to indicate the relative importance of the safety functions that a given SSC fulfils. The categories are "main" (i.e. it must be demonstrated and verified that the SSC, under normal conditions, will fulfil the required long-term safety function) or "contribute" to the fulfilment of a certain long-term safety function. The safety concept is thus constituted of the SSC and safety functions with "M" roles.

The safety concept further constitutes a framework for documenting the various safety arguments developed during establishment of the scientific assessment basis, the development of the design and the safety assessments. With this tool it is also possible to structure the remaining key uncertainties, which can e.g. be reduced by further research, development and demonstration (RDD).

Figure 2: Safety concept

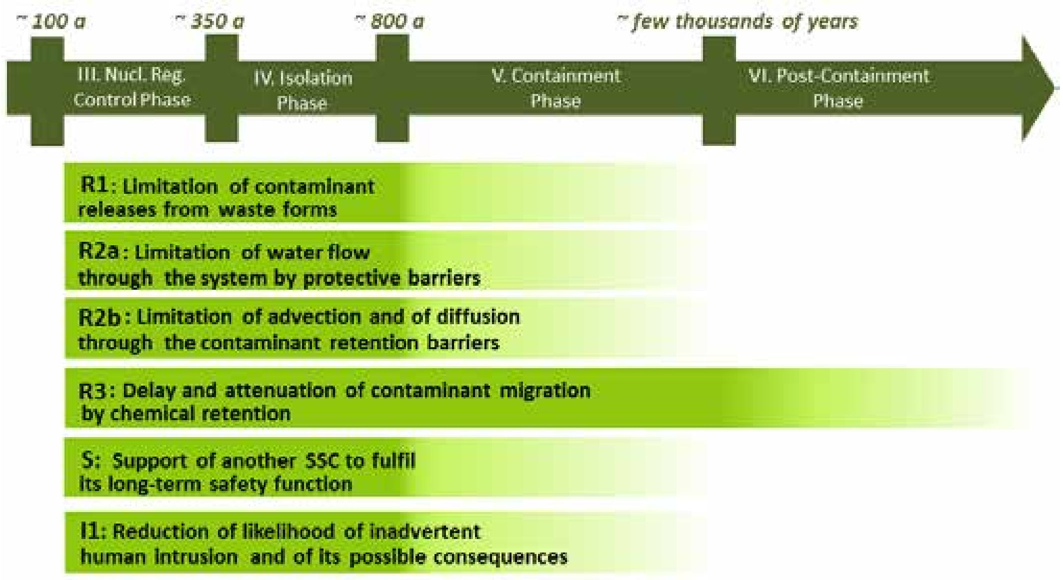


Figure 3: Principle for the development of the safety concept

	Long-Term Safety Function 1			Long-Term Safety Function 2			...
	Phase III	Phase IV	Phase V	Phase III	Phase IV	Phase V	
SSC 1	M	M	C	M	M	C	
SSC 2	M			M			
...							

Structuring the safety arguments around a central assertion

The safety arguments structured around the safety concept are part of a broader set of safety arguments (ONDRAF/NIRAS, 2012r). The safety arguments have been structured around the following central assertion:

In the current programme step of the disposal programme of Category A waste, ONDRAF/NIRAS has developed within its integrated project and its safety approach, a safety strategy and a safety concept. This strategy and concept have resulted in a design of a surface disposal facility for Category A waste at Dessel, that is optimised, of which the construction and operation are feasible and that is robust and safe. The disposal programme is ready for the next programme steps.

This assertion is supported by seven primary safety statements:

1) The current programme step of the ONDRAF/NIRAS disposal programme for Category A waste is framed in a clear decision context that successfully integrates technical and societal aspects.

2) With the integrated project and the safety approach, ONDRAF/NIRAS has developed and applied a suitable management system for the development of the safety case.
3) The safety strategy and safety concept are means to ensure a structured, clear and traceable development and documentation of the disposal and its safety argumentation.
4) The necessary knowledge base supports the assessment of facility feasibility, robustness and safety.
5) ONDRAF/NIRAS has shown that the design, construction, operation and closure of the disposal facility are radiologically optimised and feasible to implement.
6) ONDRAF/NIRAS has shown that the proposed repository is safe and robust. The assessed radiological impacts respect all safety criteria.
7) ONDRAF/NIRAS has prepared the next programme steps and has proactively started RD&D activities in order to increase the understanding of the performance of the disposal system, reduction of uncertainties, increasing confidence in the safety margins and further optimisation of the facility and its radiological contents with regard to (long-term) radiation protection and to flexibility of construction and of operation.

Each primary safety statement is supported by several secondary safety statements which are then argued with the various elements available throughout the safety case. For example:

- An important element contributing towards the integration of scientific and social aspects are the partnerships STORA and MONA that ONDRAF/NIRAS has established with the local municipalities of Dessel and Mol.
- Feasibility of construction and operation rests importantly on QA/QC programmes, a graded approach for approval from the operator and regulator for accepting waste in the repository and on the integrated management system (IMS) of ONDRAF/NIRAS (2012c). Other elements contributing to this are on-site tests such as the settlement test to verify feasibility, the construction of prototype monoliths and the demonstration test to test the constructability of modules (see Figure 5).

Figure 5: Tests to demonstrate feasibility of construction

The presentation of the safety arguments is done in parallel to a synthesis along the structure of a "safety report" (ONDRAF/NIRAS, 2012a-2012q, 2012s) that stresses more the process by which the safety case has been developed (see Figure 1):

1) establishment of context and safety strategy;
2) establishment of scientific assessment basis and freezing of data;
3) based on previous steps, establishment of the design of the disposal facility and of its implementation, i.e. construction, operation, closure, oversight (technical assessment basis) and freezing of data;
4) based on previous steps, establishment of safety assessments;
5) based on previous steps, establishment of operational conditions, i.e. waste conformity criteria, monitoring programme and technical specifications of the repository.

The combination of both types of reporting conveys a comprehensive view on the safety argumentation and its development, rather than one of the two reports taken separately.

Lessons learnt

Important challenges for surface disposal included:

1) The predominant role of engineered barriers leading to special attention towards their construction and verification.
2) The dominant role of the waste source term, leading to special attention towards the derivation of waste conformity criteria such as maximum allowable activity concentrations in the waste and maximum allowable activity in the repository as a whole.
3) The treatment in safety assessments of the evolution and related uncertainties in physical and chemical containment properties of cement-based materials (which are the predominant long-term safety barriers for surface disposal). This challenge has been tackled by a combination of conservative bounding analysis for derivation of waste conformity criteria and a set of different scenarios and assessment cases to explore uncertainties and the safety margins of the bounding analysis.

Conclusions

The license application documentation for the surface repository of Category A waste at Dessel has been established successfully along the lines of the safety case notion, initially conceived for programmes of geological disposal for medium- and high-level waste. The benefits of this have been that:

1) Safety has been incorporated in an integrated manner within all assessment basis, design and safety assessment activities.
2) The process of development of the license application has gained in clarity and traceability.
3) The license application documentation contains multiple lines of argumentation of the safety rather than argumentation based on quantitative radiological impact calculations only. To offer a comprehensive view of the safety argumentation and its development, it has been found useful to develop the safety argumentation both along a safety statements structure and along a safety report structure.

The safety case constitutes a key element of documentation supporting the license application towards a “construction and operation” license for the surface repository at Dessel. With the licensing process being started, ONDRAF/NIRAS is preparing the next programme steps of construction and operation. ONDRAF/NIRAS is currently addressing amongst other things the establishment of the QA/QC programme of cement-based materials (which are the predominant long-term safety barriers for surface disposal) and refinements related to the hydraulic conductivity of concrete, which were important topics for further work as identified by the NEA peer review (OECD/NEA, 2012).

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Development of an environmental safety case guidance manual

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Introduction

NDA RWMD is currently considering the scope, purpose and structure of a safety case manual that covers the development of nuclear operational, transport and environmental safety cases for a geological disposal facility in the United Kingdom. This paper considers the Environmental Safety Case (ESC) input into such a manual (herein referred to as the "ESC Manual"), looking at the drivers and benefits that a guidance manual in this area may provide.

Background

During a geological disposal facility development programme, from conception through closure, there are key stages where it is necessary to obtain regulatory approval (e.g. start of intrusive site investigations, start of underground operations, start of construction, start of waste disposal and start of closure operations). It is necessary to demonstrate to regulators and other stakeholders an auditable and controlled process for producing and maintaining safety cases throughout these stages.

When updating and maintaining a safety case the following topics are of interest to ensure quality and demonstrate that due process has been followed:

- the approach to addressing regulatory comments;
- dealing with updates to the safety case and change control;
- comparison of assessments over time or within a single iteration of the safety case;
- maintaining a register of uncertainties;
- demonstration that all regulatory requirements have been met.

All geological disposal projects manage such issues and it could be helpful to have procedures or guidance in place, as a means of improving understanding within the project of how these issues will be addressed and facilitating early buy-in from regulators and stakeholders on the approaches to be used. NDA RWMD is proposing to develop an "ESC Manual" as a means of achieving this. Gathering together the thinking on such topics in a single place would ensure consistency of approach, avoid *ad hoc* approaches being taken each time the ESC is updated, and provide a vehicle for communicating with regulators and stakeholders.

Scope of an ESC Manual

The idea of such a manual has been considered in the context of the UK programme where the geological disposal facility siting process is currently at a generic stage, with no specific site and therefore no information about the geology, hydrogeology or the detailed design of a future geological disposal facility. As part of the strategy for the development and delivery of safety cases to support future regulatory submissions for the site characterisation, construction, operational and closure phases of a UK geological disposal facility, NDA RWMD produced a generic ESC in 2010 for a non-site specific geological disposal facility (NDA, 2010b). This formed part of the generic Disposal System Safety Case (NDA, 2010a) which also considered transport safety and the operational safety of a geological disposal facility. The contents of the generic ESC and assessment methodologies used within it are typical of those in use by other radioactive waste management disposal organisations and are documented in the generic ESC and its supporting references.

The initial work on an ESC Manual focuses on the generic, as opposed to site-specific, ESC for the following reasons:

- Components of the generic ESC may be revised several times prior to the development of a site-specific ESC and it would be valuable for internal procedures and guidance to be in place for this in advance of any required revision. In the UK, a site-specific ESC Manual will not be required for a decade or more, allowing lessons to be learned from the application of the generic ESC Manual.
- Generic procedures and guidance that can apply to any site and disposal concept and that do not constrain NDA RWMD’s ability to innovate on a site-specific basis are likely to be the most useful and will be best able to stand the test of time. This comment applies equally well to any manual for a site-specific ESC.
- Any procedures and guidance developed for the generic ESC will almost certainly be relevant to the development of a site-specific ESC, so work done to develop procedures and guidance specifically relevant to the generic ESC will be useful not only for the generic phase, but also for the eventual development of a site-specific ESC Manual.

There may be additional components of a manual for a site-specific ESC that will need to be developed in due course, and the structure and approach developed for the manual for the generic ESC can be built on and applied in producing a manual for a site-specific ESC. However, no topics for which a new procedure would definitely be required for a site-specific ESC were identified in the course of the work underpinning this paper. Therefore, the general scope and structure of a manual for a site-specific ESC might look rather similar to that for a generic ESC.

Four tasks were undertaken to develop the scope and structure of the ESC Manual:

1) Review of progress to date on developing NDA RWMD’s Nuclear Operational Safety Manual, a parallel document that is being produced to govern the development of the Operational Safety Case. A Nuclear Operational Safety Manual is a requirement of the operational safety regulator[1] in the UK and has a scope and structure largely defined by the regulator.
2) Review of selected safety case management documentation from more advanced radioactive waste disposal projects, both in the UK for near-surface disposal and overseas for geological disposal. The UK near-surface facility safety case is of interest to NDA RWMD as it is prepared to similar regulatory guidance as that

1. This is the Office for Nuclear Regulation in the UK.

published for a UK geological disposal facility. This review was undertaken to identify relevant learning from radioactive waste disposal programmes that are at a more advanced stage of development than the NDA RWMD programme.

3) Review of selected NDA RWMD Quality Management System documents. This review was undertaken mainly to understand the extent to which the Quality Management System might already contain procedures that could serve as components within the ESC Manual, and to understand the relationship between the Quality Management System and an ESC Manual.

4) Based on the above reviews, and considering the structure and requirements of the UK's relevant regulatory guidance (EA/NIE, 2009) develop the scope, structure and potential contents of the ESC Manual.

Based on the analysis resulting from the above tasks, it is concluded that it is most beneficial for the scope of the ESC Manual for the generic phase to focus on procedures and guidance covering the following topic areas:

1) developing and updating an ESC;

2) maintenance of the generic ESC between updates.

Key topics requiring procedures or guidance for development and maintenance and change control of the generic ESC have been identified. Some of these topics sit within the wider Quality Management System as they are generic and apply across all safety cases. Other topics can form the basis for sections of a manual specific to the generic ESC. Safety case-specific topics are considered to be those where the regulators for different safety cases[2] may take a different view on how the issues should be dealt with by NDA RWMD, or where there are particular issues in the regulations that need to be addressed by development of a regulation-specific procedure.

Consideration was given to the inclusion of procedures or guidance covering safety assessment methods and default approaches and assessment parameter values in the ESC Manual, and it is concluded that:

- It is not considered necessary or useful to cover safety analysis methods, such as the approach for developing conceptual models or the approaches for deterministic or probabilistic modelling, in the ESC Manual. Reference methods are set out in NDA RWMD's 2010 generic ESC itself. Freezing such safety analysis methods in a manual could reduce flexibility and stifle innovation in the future in a field that is continuing to advance. This issue becomes even more challenging once particular candidate sites and/or disposal concepts are available and a site-specific ESC is developed. The analysis approaches will need to evolve as more information is gathered. It is therefore considered that the appropriate time to explain and agree the analysis approach is just prior to each analysis update (e.g. in an Analysis Plan, see "Analysis topics" below), and then to document the approach used in the ESC itself.

- It is considered that the specification of default parameter values for assessments should form part of a more general procedure for the specification and management of assessment parameter values covering all three safety cases, and the values themselves should not form part of any one safety case manual. Work has recently been initiated on an NDA RWMD-wide framework and tools to help manage assessment parameter values and to ensure that parameter values common to more than one safety case are used consistently.

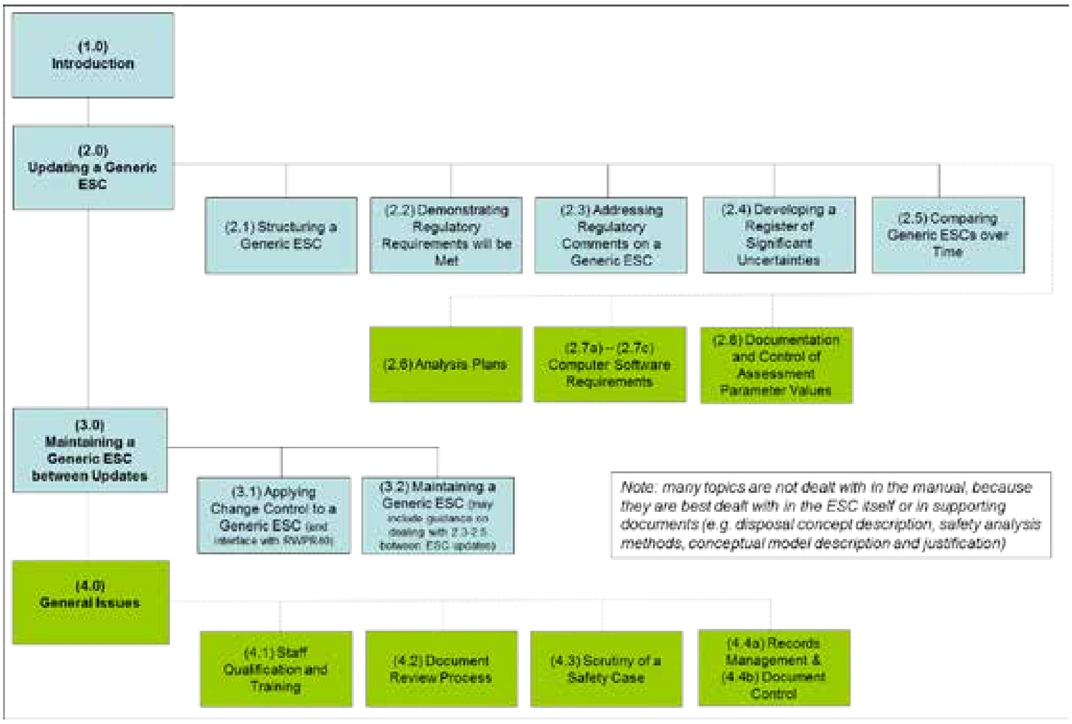
2. For a geological disposal facility in the UK, the different safety cases here refer to the areas of transport, operational and environmental safety for which different regulators (or departments within regulatory bodies) are responsible.

Proposed scope/structure of safety case manual topics relevant to a generic ESC

The proposed structure of the ESC Manual is illustrated in Figure 1.

Figure 1: Proposed scope and structure of safety case manual topics relevant to a generic ESC

Components that could be specific to the manual for a generic ESC are shaded light blue. Components that should not necessarily differ between safety cases and that are assumed to form procedures within the wider QMS are shaded light green.



The generic topics that have been identified as relevant to the development and maintenance of the ESC include general topics and topics related specifically to the conduct of safety assessments. In general, it is likely to be preferable for these topics to form part of the wider Quality Management System, such that they will only need to be dealt with in the manual if not adequately covered in the wider Quality Management System or if more specific or detailed guidance is required for a particular safety case. These include:

(2.6-2.8) Analysis topics

- Analysis plans prior to conducting any significant modelling studies[3] that are expected to generate results to be used to make design, analytical, operational or regulatory compliance decisions.
- Computer software requirements, including computer code verification, conduct of routine calculations and code configuration management.

3. Analysis plans could also be important for design, experimental and site characterisation activities.

- Documentation and control of assessment parameter values.

(4.0) General topics[4]

- Staff qualification and training.
- Internal review by NDA RWMD staff and contractors of quality-affecting documents (referred to as "document review" here).
- Obtaining external review and NDA RWMD management approval of the ESC and other safety cases (referred to as "scrutiny" here).
- Records management and document control.

Work is already under way within NDA RWMD to refine or develop procedures or guidance for some of these topic areas. There is also experience to draw on from other projects that have developed procedures covering these topics.

Topics for which procedures/guidance could be prepared specifically for the ESC

The following structure and initial contents list for the ESC Manual (generic phase) is proposed:

Section 1: Introduction

Section 2: Updating a generic ESC

(2.1) Structuring a generic safety case (ESC)

Consideration should be given to the structure of the ESC main report, and the entire ESC document suite, prior to each update of the ESC. Consideration should be given to such things as the structure of the previous version of the ESC, any comments on that structure, new international or UK regulatory guidance on structuring an ESC, the purpose and scope of the ESC update, and any other new information that has emerged that may be relevant.

(2.2) Demonstrating how regulatory requirements are met or will be met

It could be argued that regulatory submissions (ESC and other environmental permitting documents) should be aligned with the structure of the regulatory guidance, to ease the task of the environmental regulator[5] in evaluating the submissions. However, this may not be appropriate for updates to the generic ESC, because it has additional purposes than addressing compliance, because NDA RWMD will not want to be constrained in determining a structure that communicates its safety case, and because the regulatory guidance will be periodically revised. The 2010 generic ESC (NDA, 2010b) includes a cross-walk identifying where in the project documentation each requirement and/or paragraph in the regulatory guidance (EA/NIE, 2009) had been considered or met. The cross-walk can be updated iteratively over the development cycle for an ESC, so that it is used initially as a tool to help identify gaps in documentation and then in the ESC update to demonstrate which requirements the ESC is addressing and where in the documentation each requirement is addressed. Guidance in the manual could set out the format of a cross-walk and the level of detail.

4. Note that this is not a complete listing of requirements for general programmatic topics.
5. This is the Environment Agency in the UK.

(2.3) Addressing regulatory comments (on the generic ESC)

Guidance in the manual could help ensure a common approach was followed across the project, and that there was regulatory buy-in of the approach.

(2.4) Developing a register of significant uncertainties (ESC register)

In the UK, the requirement to establish and maintain “a register of significant uncertainties” is discussed by the environmental regulator’s guidance (EA/NIE, 2009). NDA RWMD’s approach to managing uncertainties is set out in the 2010 generic ESC, and an initial register appropriate to the current generic programme stage is provided there. However, there is no unique approach to developing and maintaining such a register. Guidance in this area would help ensure consistency in vision and approach going forward. This could include a definition of the term “significant” with respect to the generic ESC and could set out the form of the register and could provide a link to change control that might be required when significant uncertainties are addressed.

(2.5) Comparing safety cases over time (ESC)

When ESC and their supporting assessments change over time – as they most certainly will – NDA RWMD should be able to provide a clear rationale for why the ESC has changed and the implications of the change, and the changes in results should be possible to trace through to changed or new assessment approaches, scenarios, modelling assumptions, codes, and/or parameter values linked to new information. The development of guidance specifically covering this topic could help ensure that appropriate information is provided in updates to assessment documents and the ESC main report.

Section 3: Maintaining a generic ESC between updates

The following topics have been identified that relate to ongoing maintenance of an ESC between updates:

(3.1) Applying change control to an ESC

Guidance could help identify how change management should be applied within the ESC itself. The kinds of things where change control would need to be applied for the generic ESC include the use of new assessment codes or analysis methods, or changes that need to be made to assessments in response to new information (e.g. inventory changes, improvements in the understanding of physical and chemical processes). The ESC-specific procedure could consider the possibility of developing a system to trace links between assessment “components”, e.g. between features, events and processes (FEP), conceptual models, codes and parameter values, in order to ensure that appropriate consideration is given to the potential knock-on impacts of making a change to one part of the assessment, on related parts of the assessment.

(3.2) Maintaining a generic ESC

A range of questions needs to be addressed concerning how an ESC will be maintained between updates and, particularly for the generic ESC, to identify at what stage it would be appropriate to re-release ESC documentation. Consideration needs to be given to:

- What kind of a change or new information external or internal to the generic ESC could lead to the need for an update? For example, one possible reason to update the generic ESC could be if there were to be a significant change in regulatory guidance.
- How should inventory changes be dealt with between ESC updates?

- What should be done if comments on an ESC update are addressed between updates, or if there is a change to the register of significant uncertainties (see manual sections 2.3-2.5)
- What should be done about correcting minor errors or other minor changes between ESC updates?

Summary

In summary, NDA RWMD is currently considering the potential scope, purpose and structure of a safety case manual that covers nuclear operational, transport and environmental safety case development. This paper considers those components relevant to producing a guidance manual to provide information on how to maintain and update a generic ESC for a geological disposal facility and a proposed scope and structure for NDA RWMD's context has been outlined. The topics covered represent the current input from an environmental safety case perspective into development of a safety case manual, which is currently a work in progress and therefore not a finalised company position. However, NDA RWMD believes that developing environmental safety case components of a safety case manual could be helpful as a means of improving understanding within the project of how the generic-phase safety case will be maintained and updated. Gathering together the thinking on such topics in a single place would ensure consistency of approach, avoid *ad hoc* approaches being taken for each update, and provide a vehicle for communicating with regulators and stakeholders. Furthermore, the process of developing a safety manual can help to identify areas for improvement within existing quality management systems, identify future needs for such systems and ensure a common understanding of the structure and scope of the safety case.

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Licensing review process of the European Spallation Source (ESS) research facility

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Introduction

On 3 January 2012 a license application under the Radiation Protection Act (SFS, 1988b) for the European Spallation Source research facility was submitted to the Swedish Radiation Safety Authority. The European Spallation Source research facility will be the site of a new and quite unusual kind of neutron source, based on a large proton accelerator that bombards a heavy material with protons. The Swedish Radiation Safety Authority is now reviewing the application.

The European Spallation Source (ESS)

The proposed European Spallation Source (ESS) is to be a multi-scientific research facility which might be described as an enormous microscope for examining molecular structures. The ESS will generate a very powerful long-pulse source of low-energy neutrons (rated at 5 MW) 30 times more intense than achieved with similar sources in operation today. These neutron beams will enable scientists to analyse and understand basic atomic structures and forces of a variety of different materials. There are applications in disciplines such as medicine, chemistry, physics and engineering.

Today, 17 European countries are engaged in the ESS project, with Sweden and Denmark acting as host nations for the facility. The Swedish company European Spallation Source ESS AB (ESS AB) was established in 2010 with the aim of constructing and operating the ESS facility. The company's main shareholder is the Swedish government (75%) and the Danish government is the main co-owner (25%). According to ESS AB's current plan the ESS facility should be in operation in 2025. It is envisaged that the ESS will be in operation for about 40 years.

The ESS research facility will be located in southwest Sweden, on the outskirts of the city of Lund. In addition, a supercomputing data management and software development centre will be located in Copenhagen. It is expected that 2 000-3 000 guest researchers will carry out experiments at ESS each year. Easy access to the facility and to the data produced for university and research laboratory users all over Europe will be a high priority, just as maximising industrial access will be.

The design and construction of the ESS facility includes three main components: a superconducting particle accelerator, about 580 metres long, where protons are accelerated using high-voltage equipment (klystrons) and cryogenically cooled equipment; a target station where the protons hit a target material (tungsten) leading to generation of neutrons; and, finally, the instruments located at different distances from the target (15-300 metres) to which the neutrons are guided in tubes to the samples surrounded by investigation detectors. The ESS facility will not be a nuclear facility, but it will house considerable quantities of radioactive material on par with that of a research reactor. More information on the ESS facility can be found in the ESS Technical Design Report (ESS, 2013).

In conjunction with ESS, the Max IV Laboratory (MAX IV) is being constructed. MAX IV will be a synchrotron light facility for studies of molecules and atoms. MAX IV is to be run under the auspices of Lund University. The layout of the facility surrounds a big electron accelerator. From the perspective of radiation safety, the risks and problems posed are smaller for MAX IV than those related to ESS. The application for MAX IV was submitted in 2011 and will be reviewed by SSM in co-ordination with the ESS application.

Figure 1: A conceptual model of the ESS research facility

Source: Used with permission from ESS AB.

Radiation generated by ESS

The ESS research facility must have protective barriers in order to shield the surroundings against radiation. The source of radioactivity is to be equipped with radiation absorbers that reduce radiation as well as containment systems that prevent radioactive substances from spreading to the surroundings. When the accelerator produces a high-energy proton beam, this also involuntarily generates neutron and gamma radiation. Direct prompt radiation is only generated when the accelerator is running. In addition, radioactivity is generated in the materials hit by high-energy direct radiation. The highest level of radioactivity is generated in the target, but also to a lesser extent elsewhere, such as the air and soil surrounding the accelerator.

The review process

SSM has drawn up routines for the review process and preparation of licenses concerning nuclear facilities and other complex facilities/activities involving ionising radiation. The review process for the ESS application is in compliance with these routines and will take place in several steps according to the following:

1) A license is issued for activities involving ionising radiation.

2) A license is issued for commencement of the construction phase (installation).

3) A license is issued for test operation.

4) A license is issued for regular operation.

5) A license is issued for decommissioning.

ESS AB submitted a license application under the Radiation Protection Act (SFS, 1988b) for the ESS research facility to SSM on 3 January 2012. On 15 March 2012 ESS AB supplemented the application with a preliminary safety analysis report, PSAR (ESS, 2012b) and an environmental impact assessment (EIA) (ESS, 2012a). Thus the first step of the review process could be initiated. Also in March 2012, ESS AB submitted an application under the Swedish Environmental Code (GOS, 1988) to the Land and Environmental Court.

When the application from ESS AB was received, the first thing done was an initial acceptance review. The aim was to conduct a general assessment regarding the completeness of the application documents. This initial acceptance review took around three months, during which time SSM consulted on the ESS application with various stakeholders nationally and also in Denmark.

The application was viewed as being insufficiently complete to enable SSM to begin its technical review work. On 26 July 2012 SSM requested ESS AB to supplement the application with documents concerning most aspects of the construction: preliminary risk analysis, construction of structures and facilities, technical system design, radioactive waste management, decommissioning, radiation protection of workers, financial liabilities, the EIA, etc.

By 31 December 2012 SSM had received some of the required supplements from ESS AB. Still pending are documents on radioactive waste management, decommissioning and financial liabilities. According to ESS AB these last supplements will be delivered at the end of March 2013.

In April 2013 the technical review work is planned to start, where the first phase is a broad review of the application to determine whether the application is now sufficiently complete and of sufficient quality to enter the extensive main review phase. SSM anticipates that this first phase is finished by the end of April 2013 and that the main review can begin after that.

As mentioned above, the licensing of the ESS facility is also done in accordance with the Swedish Environmental Code, handled by the Swedish Land and Environmental Court. However, by praxis in Sweden, SSM will handle the regulatory questions regarding radiation safety and radioactive waste. The environmental impact assessment will thus be reviewed and considered by both SSM and the Land and Environmental Court.

In addition to the two license review processes under the Radiation Protection Act and the Environmental Code, the Swedish government might also consider the permissibility of the ESS facility. Whether this will be the case or not remains to be decided.

Legal requirements

SSM will examine the ESS application for compliance with current legislation, which in this case will be the Radiation Protection Act and the Radiation Protection Ordinance (SFS, 1988a, 1988b). The main purpose of the Radiation Protection Act and Ordinance is to protect people, animals and the environment from the harmful effects of radiation.

In addition to the Radiation Protection Act and Ordinance, SSM has issued a large number of regulations supplementing the Act and Ordinance. Due to the unique character of the ESS facility – it is not a nuclear facility but it will house considerable quantities of radioactive material and wastes on par with a research reactor – not many of the existing SSM radiation protection regulations are directly applicable. The regulations that are applicable are the following:

- the Swedish Radiation Safety Authority's regulations on operation of accelerators and sealed radiation sources (SRSA, 2008c);
- the Swedish Radiation Safety Authority's regulations concerning basic provisions for the protection of workers and the general public in practices involving ionising radiation (SRSA, 2008a);
- the Swedish Radiation Safety Authority's regulations on external workers in practices involving ionising radiation (SRSA, 2008b).

SSM has identified a need for additional requirements. These additional requirements will be derived from the applicable aspects of SSM's regulations on nuclear facilities even though the Act on Nuclear Activities (SFS, 1984) is, in terms of legislation, not directly applicable to non-nuclear facilities.

ESS AB has stated that they will base their safety on among others the IAEA Safety Standards Safety of Nuclear Power Plants: Design, Requirements (IAEA, 2000) and the Safety Assessment for Facilities and Activities (IAEA, 2009).

Some key issues

For SSM, a number of key issues and concerns have arisen in connection with the review process of the ESS application. A selection of key issues is presented here.

When preparing for the review process of the ESS application, SSM was apprehensive about not covering all aspects of the rather complex facility that ESS will be. During this stage SSM went on a number of study visits to accelerator facilities in different parts of the world, to learn more about how the licensing procedures were performed in each case and to develop a better understanding for the kind of radiation protection issues occurring at this type of facility, among other things.

The first step in the review process is of key significance, as several important decisions are to be taken, for example relating to the selection of site. The chosen site for the ESS facility is on the outskirts of Lund, which is a rapidly expanding city. SSM has to ensure that, from a radiation safety point of view, the facility can be built at the suggested site.

Even though SSM should issue ESS AB a license for the ESS facility at Step 1, the process will not end here. The first step of the process, together with the following steps, will take several years. It is inevitable that a number of experts involved in the reviewing at SSM will leave and have to be replaced at intervals, which might pose problems regarding the continuum and efficiency of the review process.

SSM will review ESS AB's application and determine whether statutory requirements imposed on radiation safety are fulfilled. However, at this stage ESS AB cannot present all details in terms of construction and operation of the ESS facility, as it is still a conceptual design. ESS AB therefore must demonstrate at this step (and the following) that they are capable of constructing and running the facility in fulfilment of the requirements imposed. For SSM, it might be a challenge to decide what needs to be demonstrated by ESS AB at this stage, in order for the authority to feel confident that ESS AB has prerequisites to fulfil the requirements later on when the ESS is being constructed and in operation.

Significant volumes of radioactive waste will be generated at the ESS facility. The non-nuclear industry, hospitals and research centres in Sweden rely on the system for radioactive waste management created by the nuclear industry: the radioactive waste is sent to Studsvik Nuclear AB (SNAB), the only approved radioactive waste management facility in Sweden. After treatment, the radioactive waste is stored by SNAB, pending disposal in either SFR, the repository for short-lived low- and intermediate-level waste, or in SFL, the planned repository for long-lived low- and intermediate-level waste, which will be in operation in 2045 at the earliest. SFR and SFL are owned by the Swedish Nuclear

Fuel and Waste Management Co (SKB AB). SNAB has an agreement with SKB for the disposal of radioactive non-nuclear waste in SFR. SNAB and SKB AB are currently discussing a letter of intent for the disposal of radioactive non-nuclear waste in SFL. ESS AB is currently involved in a dialogue with both SNAB and SKB AB on how to manage the radioactive wastes generated by the ESS facility; it is not yet known if it will be possible to use the radioactive waste management system created by the nuclear industry for disposal of the ESS wastes.

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Radioactive and conventional toxic waste compared – An integrated approach, useful for an appraisal of carbon capture and storage (CCS)

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Goal

The interplay of nuclear and conventional toxic ("special") waste is investigated, using a novel integrated system assessment: material and system characteristics, risk assessment and regulatory approaches. The goal is to create profiles of strengths and weaknesses of wastes that are similar in their risk characteristics but dealt with differently in risk management and regulation. A further objective is to draw lessons from the comparison of different discourses and procedures of waste with a similar profile with regard to decision-making processes (the reasons for the different regulation of both waste systems are not investigated here). Finally, a side glance is ventured on carbon capture and storage (CCS) in view of the keynote lecture of Session 5.

Background

The different perception and management of comparable technical risks have long been explored by various sciences, and attempts for harmonisation were proposed (Flüeler and Seiler, 2003), including an appraisal of both waste fields (Brasser, 1995; Little, 1996; Flüeler and van Dorp, 2000). The present assessment approach was successively developed – conceptually (Flüeler, 2011), methodologically and substantively (Flüeler, 2006, 212a). This contribution is based on Flüeler (2012a, 2013). The focus of that work lay on technical and procedural aspects supposedly relevant for the perception by various actors. The structure followed four steps: *A)* analytical fundamentals and system inter-relationships; *B)* decision-making processes in waste management; *C)* interaction of A and B; D) suggestions for further research: Formulation of hypotheses and scenarios for the study on perception (Part II, not covered in this contribution). Here we present central findings of each aspect or criterion under scrutiny (we are aware that our overview neglects possible national cases and is intrinsically generic, the empirical system basis is predominantly Switzerland).

System characteristics

On far as the waste notion, there is no "scientific" or even object-based definition of what "waste" actually is, neither in the nuclear nor the special waste field. What represents "waste" depends on its user or owner and is, therefore, a social construct. The boundaries between resources, products and waste are blurred. Thus, decision strategies and regulations cannot be separated from the (objective) respective waste system. In the conventional field, all substances have to be investigated with respect to their classification (as resource or waste, e.g. batteries may have a positive market value if recycled, negative if they are a legacy to be landfilled; mercury can be reused as an extraction medium in gold mines or emitted to the environment, it has been valuable in clinical thermometers or batteries, it is, however, going to be banned or extensively reduced shortly (UNEP, n.d.).

Substances burdened with radioactivity are cleared or decontaminated if possible. The remaining waste is classified into around 160 types for conditioning and disposal (Nagra, 2002, 2008). Conventional materials can either be completely recycled, burnt, conditioned or landfilled (in which case they are indeed considered waste). Special waste in particular can be divided into around 450 waste types, of which one-fifth (in Switzerland) is disposed of in a dedicated landfill (BAFU, 2012a). For the sake of completeness, mixed waste also exists, meaning hazardous waste with radioactivity.

Regarding the management principles, sustainability suggests to close material cycles in both fields. The ultimate closing of the fuel cycle has been the strategy in the nuclear industry ever since the OECD/NEA report *Nuclear Energy in a Sustainable Development Perspective* (2000) was issued (even though current reprocessing extracting plutonium, partly uranium, for reuse, cannot be considered full recycling). Once waste is defined, it must be kept away from man and the environment by disposal in landfills or repositories. In both fields the principle of “confine and concentrate” has finally prevailed over “dilute and disperse” even though in reality this is not always the case, and exceptions do exist (e.g. the trade-off in ^{14}C treatment between real doses today and potential doses in the future; or natural microbial decomposition versus a rigorous concentration strategy). Both waste communities agree that waste is either a resource in the wrong place (not yet recycled) or residual material to be disposed of. Residuals are toxic in the long term and require safe long-term disposal.

Long-term deposits are usually above ground in the conventional domain, and are often (always in the case of high-level waste) in deep geological repositories underground in the nuclear domain. Both fields exhibit areas of transition, such as controllability and retrievability on the one hand, no need for maintenance and ultimacy (in the sense of no intention of retrieval) on the other hand. Therefore, target conflicts are quite possible, as can be shown with the notion of retrievability, either meant in case of system failure or to have access to valuable material (Flüeler, 2012b). Whereas in the radioactive field there are basically only two (or three) repository types (for low-level and high-level waste), the conventional field maintains a lively dynamic, partly by increasing deposition, and partly through progress in recycling. Thus, there exists no accurate selectivity as to which “special waste” will be effectively deposited in the future or rather recovered.

With respect to quantity (not toxicity) it is quite obvious who prevails. In Switzerland only, the annual accumulated special waste to be deposited in landfills corresponds to the total waste the five nuclear power plants produce during an operation period of 50 years (including an expected nuclear waste from medicine, industry and research over a collection period of 80 years). In the conventional sector, however, the recycling rate is high and increasing, whereas the nuclear handling is limited to decontamination or possibly reprocessing (BAFU, 2012a; Nagra, 2002).

The conventional domain is characterised by a great number of actors and a multitude of regional regulatory bodies, whereas producers and users of nuclear material are small in numbers, the competences are centralised and supervision rests with a few federal agencies. Even though there is a flourishing international waste business, the conventional community is usually regionally, at best nationally, connected regarding municipal solid waste and sewage sludge; the nuclear community, though, is closely internationally linked and institutionalised (with standardisation and review mechanisms).

By law financing both fields of application is, in general, based on the polluter pays principle (PPP, or principle of causality). It is evident, however, that this is easier to follow in the nuclear field with relatively few actors and waste types (and a dedicated up-front producer-paid fund for disposal) whereas the picture is much more complex in the case of conventional waste management. With municipal waste the Swiss Federal Supreme Court ruled that all municipalities had to introduce a so-called bag fee; not more than 30% of the disposal costs are allowed to be paid by public funds, i.e. taxes. This PPP conforming sentence results in about one-third less municipal waste, and this fraction is collected separately and largely recycled (BAFU, 2012b). Waste from the building industry and trade are organised by the private sector itself. Sewage sludges are paid by way of a domestic and industrial waste water fee, respectively. Special waste is either covered by a pre-paid disposal fee or the waste specialist directly charges a fee to the waste owner. Long-term safety in the waste treatment on landfills, however, is not covered by the waste producers but by the state. Concerning legacy waste (on contaminated sites) in Switzerland there exists a separate fund provisioned by the respective land owners or, in case they cannot be held accountable, the state.

Nuclear repositories and conventional landfills are both associated with a similar risk mechanism: a long-term chronic release of the pollutants into the environment. The performance assessment for nuclear waste is not addressed here (OECD/NEA, 2008). The comparable complexity of the disposition of special waste is not reflected in the practised risk methodology. In this domain we are also confronted with time frames of hundreds to hundreds of thousands of years. Swiss standard leaching tests, even though stringent in an international comparison, permit no information on the long-term behaviour of a disposition system, let alone long-term safety (SFA, 2011). The definition of protection goals for landfills is vague and quite general, whereas the nuclear domain works with calculable annual doses which may be compared with a defined regulatory limit as the protection goal. The measured variable here is radioactivity, which can usually be accurately quantified whereas the conventional domain largely refers to concentrations of contaminants. The heterogeneous composition of waste increases the risk to miss interactions of known and even new substances.

Apart from unintentional human intrusion, the main risk mechanism of repositories and landfills (and contaminated sites, for that matter) is a low-level but long-term chronic release into the environment; it may be described as a slow degradation of an open system (geological formations and geotechnical barriers/weather conditions, respectively) with concurrent large uncertainties. Substances with chronic effects are hard to detect. In the nuclear field, they are quantified by means of an uptick of radioactivity; in the conventional field, however, concentrations of noxa are given in the absence of impact analyses.

In view of the recognised complex system mechanism there is consensus in the nuclear community that for the required long-term safety of repositories "is not intended to imply a rigorous proof of safety, in a mathematical sense, but rather a convincing set of arguments that support a case for safety" (OECD/NEA, 1999; OECD/NEA, *et al.*, 1991, *passim*). In the conventional field, understanding of the (technical) system and the data basis is often insufficient. Input data (e.g. chemical-physical waste characteristics), but even more so long-term mechanisms (e.g. multiple-phase thermodynamics), are not well known. Flux and transport models in unsaturated conditions are premature and partly difficult to obtain. The rough understanding of individual processes, to date, does not permit reliable safety analyses for above-ground disposition systems (Herrmann, 1998, *passim*).

Processes and procedures

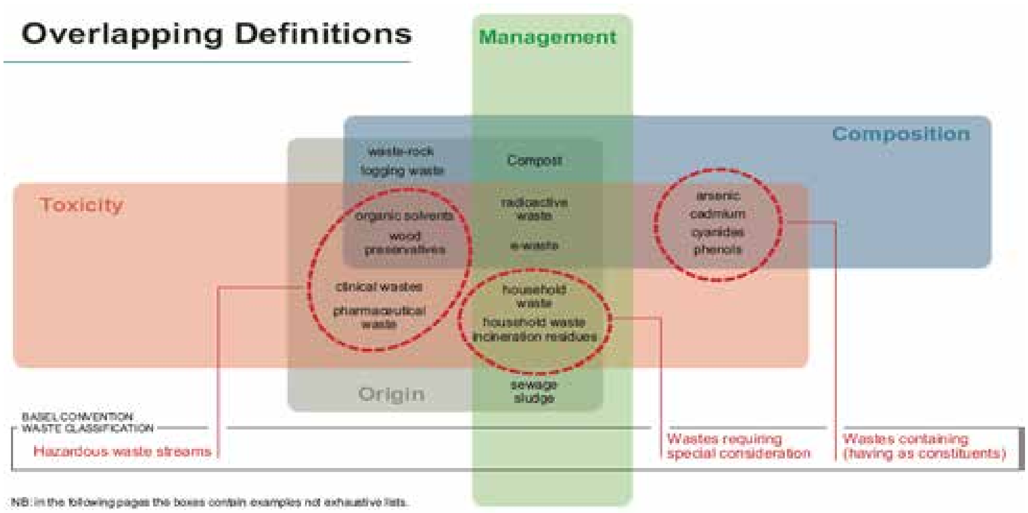
As "waste" cannot be scientifically defined and is a societal construct the associated processes (of handling, site selection and decisions in general) are pivotal. In both fields under investigation there are major failure stories, albeit not rendered in this presentation (nuclear, e.g. Asse in Germany, conventional, e.g. the hazardous waste dump of Kölliken in Switzerland – both are cases with a clear remediation need). Systematic site selection approaches have gained a foothold in the nuclear domain (e.g. BFE (2008) following and adapting the blueprint of AkEnd (2002), see Session 3.2). The conventional field offers the advantage of having a large option space for most conventional hazardous substances.

Some conclusions for further studies of integration

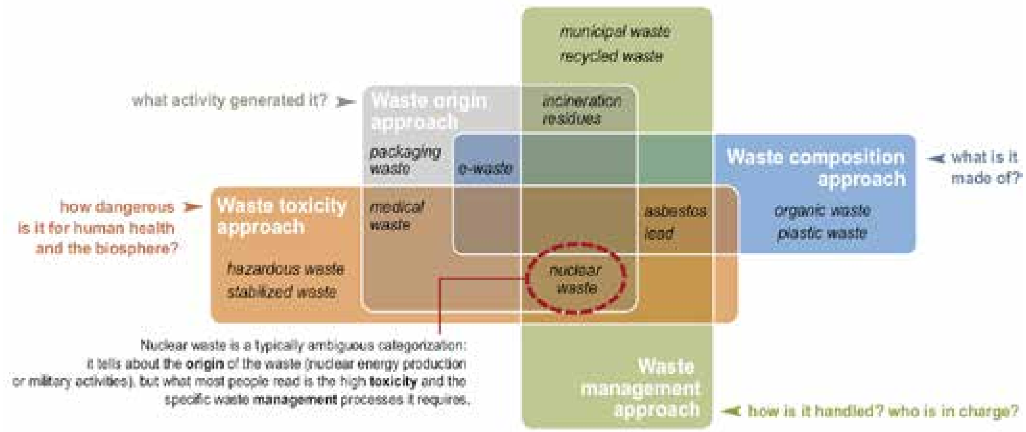
Science cannot define what waste is, in either field. Thus, decision strategies and regulation cannot be separated from the "objective" waste system (Figure 1).

Figure 1: Waste substances of both domains are in a continuum even in their "technical" definition, but also depending on the focus selected: composition, hazard potential, origin (polluters) or handling

a) Conventional domain



b) Nuclear domain



Source: UNEP (2004), p. 7.

In the conventional domain, all substances have to be investigated in view of their classification (as resource or waste). Waste is a resource in the "wrong" place or a residual to be disposed of. Residual substances are toxic in the long term and have to be disposed of accordingly. Procedural and process issues make it explicit that perspectives (and their changes) and dynamics play an important role. A clear regulation of parameters and requirements as well as competencies and responsibilities of the involved parties are central. Independent regulators, on a par with the implementer, must accompany and supervise all relevant steps. They need respective competence, resources and staff. It is wise to adequately involve third parties – and the larger public – at an early stage.

A nuclear-waste-informed look on CCS

Carbon dioxide (CO_2) capture and storage (CCS) is a technical option for avoiding higher CO_2 concentrations in the Earth's atmosphere while still using carbon-intensive technologies in power production and other industries. It highlights the tension between the advantage of a short-term "quick fix" and the disadvantage posed by the risk of long-term leakage and, from a technology policy perspective, the danger of perpetuating carbon lock-in. In another study (Flüeler, 2012a) we assessed CCS against criteria taken from the controversial and long-lasting governance of radioactive waste (recognising that CO_2 is not a conventional toxin but harmful in the context of climate change). As the dimensions of this issue are manifold and intertwined, there is no "one" methodology with which to analyse it (such as a technology assessment of the n-th order). Instead, cross-disciplinary investigations make it possible to draw lessons from contentious long-term environmental issues and social science research, which is necessary before embarking on this route on a large scale.

The approach was to assess the CCS issue using a mix of disciplines and perspectives. These can include systems theory, integrated risk assessment, social sciences, technology assessment and management (implementation, compliance) (Flüeler, 2006). The following six criteria address issues that have proven to be crucial in technology policy debates [(Ropohl, 1999; Ravetz, 1980; Vlek, 1980; Wynne, 1983; Morone, 1986; Kasperson, 1992), approach developed in Flüeler, 2006)]: *i)* need for deployment and comparative benefits; *ii)* total system analysis and safety concept; *iii)* dedicated, internationally harmonised regulation and control; *iv)* economic aspects (costs and incentives); *v)* implementation; *vi)* societal issues. Here we focus on the safety concept in No. 2.

Comparing radioactive wastes with CO_2 reveals differences and similarities, both in systemic and risk aspects (Tables 1 and 2, respectively). It is less about the sheer size of the release into the environment and has more to do with the nature of it (Benson, 2005). Disposal and storage systems are both associated with a similar risk mechanism: It is expected that, just as with toxic waste above, a low-level but long-term chronic release of pollutants into the environment will happen, along with a slow degradation/alteration of an open system (geological formations and geological formations/climate, respectively) with concurrent large uncertainties. With the exception of some scenarios of human intrusion, potential impacts are hard to detect with respect to location and time.

In the radioactive waste community, evidence, analysis and arguments for the safety of radioactive waste disposal sites must be gathered into a so-called "safety case", which involves a stepwise and iterative procedure to provide risk assessments and appraisals, as well as confidence statements from site selection to closure and post-operational monitoring of a specific facility (OECD/NEA, 2008). The key idea of compliance is the multiple-barrier concept, according to which a multitude of technical and natural barriers take effect in case one single barrier breaches (though no complete redundancy can be achieved).

This defence-in-depth paradigm, which is central to nuclear installations, is missing in the CO_2 safety concept for geological storage (IPPC, 2005, p. 35). The basic idea here is that caprocks, such as aquifers, must prevent CO_2 from leaking back through sealed, abandoned or dedicated injection wells or other pathways into the atmosphere. The dependence on what is basically one barrier is risky even if potential storage locations, such as (relatively few) depleted oil and gas reservoirs, are proven long-term traps of gas.

It is symptomatic that the term used is "storage" rather than "disposal". In the nuclear community, the term "storage" always refers to interim storage, whereas disposal is meant to be final in the sense that there is no intention of retrieval and that, if successful, (most) radionuclides will be kept out of the biosphere (IAEA, 2006, p. 1). Retention is also possible by building CO_2 into minerals (mineralisation, carbonation) but this has only been conceptualised in research, let alone implemented as state of the art (Kelemen, 2008).

The CCS community is far from an intricate build-up of scientific evidence and confidence, even though the issue has been recognised and a degree of international harmonisation is under way (IEA Greenhouse Gas R&D Programme) and science has become institutionalised [for example, by creating a dedicated journal, the *International Journal of Greenhouse Gas Control* (Gale, 2007)]. The needed technical system goes beyond the admittedly well-known injection of CO_2 but must still encompass the aspects described in the nuclear waste safety case approach. This means that CCS technology is not yet "mature", as maintained earlier (Bachu, 2000, p. 957). Maul, *et al.* (2007) recognise the following research demands in CCS:

> *Dealing with the various types of uncertainty, using systematic methodologies to ensure an auditable and transparent assessment process, developing whole system models and gaining confidence to model the long-term system evolution by considering information from natural systems. An important area of data shortage remains the potential impacts on humans and ecosystems.* (p. 444)

Grünwald (2008) and Jacobson (2009) list the research demands on leakage phenomena (nature, timing, mechanisms, etc.), while Savage, *et al.* (2004) add scenario development to the agenda.

Table 1: System and institutional properties of radioactive waste disposal and fossil CO_2 storage

The large family of technologies is an asset of CO_2 whereas the absence of economic incentives and disposal funds, the high number of actors (polluters and potential implementers), the uncertain regulation and classification, the high volume and the non-dedicated type of facilities are burdens to sustainable CO_2 storage

Characteristic \ System	Radioactive waste	(Fossil) CO_2
Origin	Nuclear electricity generation; medicine, industry, research	Fossil-fuelled energy generation; oil refinery; cement, iron, steel industries
Sources	Comparatively few	Few large, innumerable small
Economy	Polluter-pays fee on each kWh	Not regulated to date
Resources (R&D, money)	Substantive	Sparse
Actors	Few state-by-state implementers; national regulatory agencies	Market-driven international companies; regulation undetermined
Volume (world wide)	Small (Mt/60 y)	High (Gt/y)
Waste package	Concentrated, consolidated waste stream	Dispersed
Nature of substance	Contaminant, waste	Greenhouse gas, commodity (waste)
Regulatory classification	Internationally classified	No standardised classification
Facilities	Dedicated repository with drifted galleries, abandoned mines substandard	Exploited reservoirs
Range of options/flexibility	Small	Large [from alternatives (e.g. renewables) to variants (e.g. ocean storage, mineralisation, etc.)]

Source: Flüeler (2012a), p. 204.

Table 2: Risk-related properties of radioactive waste disposal and fossil CO_2 storage

The longevity of pollutants, leakage as evasion mechanism, the low but chronic risk level and the long-term barrier mechanism are common characteristics. The relatively simple hazard system, good traceability and the wide range of options may be assets of CO_2, whereas most other waste and risk management aspects are at an initial phase and are therefore currently burdens to sustainable CO_2 storage.

Characteristic	Radioactive waste	(Fossil) CO_2
Pollutants		
Waste package	Concentrated, consolidated waste stream	None, dispersed
Toxicity	High (and low with short-lived low-level waste)	Low but dispersed
Duration of toxicity (residence time in systems)	Very long (millions of years)	Long (thousands of years)
(Long-term) hazard system	Complex (ionising radiation)	Simple (greenhouse gas effect)
Risk level	Low, chronic	Low, chronic
Protection limits	Radiation dose (mSv)	None (work safety: yes)
Indicators	Defined for safety functions (such as isolation, confinement, decay)	None (no thresholds)
Traceability	Bad	Better
Transport mechanism	Groundwater, air (gas)	Air, groundwater
Evasion mechanism	Leakage	Leakage, seepage
Risk appraisal		
Concept	Concentrate and confine	Confine
Barrier mechanism	Sorption, diffusion	Mineralisation, dispersion
System understanding	Existent, developed	Under construction
Site characterisation	Good, dedicated investigations (seismics, boreholes, etc.)	Medium (lost records of abandoned reservoirs)
Barriers: Engineered (technical) Natural	 Pellet, canister/container, bentonite Sealed galleries and shafts Host rock and other confining geological units	 None Sealed wells Cap rock(s)
Assessment methodology	Relatively well-established performance assessment ("safety case")	Embryonic
Other relevant issues		
Conflicts in land use	Partly existent (can be avoided)	Often intrinsic (abandoned resource reservoirs)
Process of demonstration of safety	Stepwise, iterative	Undefined
In situ monitoring	Accessible shafts and galleries	Basically inaccessible (single-barrier seals)
Range of options/flexibility	Small	Large (variants, e.g. ocean storage, mineralisation)

Source: Flüeler (2012a), p. 205.

The institutional situation in the CCS field (involving unclear regulation, many global players, and lacking financial incentives and funds) means that no standard site selection process has been developed. Despite this, many individual and detached projects have been pursued to date. The diverging degree of detail can be seen from Figure 2 in general or in applications (Ramírez, 2008; Chalmers, 2009; Jacobs, 2009), as opposed to the site selection procedure for radioactive waste in general (OECD/NEA, 2004), or as proposed or applied in a national programme (AkEnd, 2002; BFE, 2008). The above conclusions from the comparison with conventional toxic waste also hold true for CCS.

Figure 2: Phases and permits typically associated with CO_2 storage projects

Source: Aarnes (2008), p. 1736.

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Session 8

Societal Context of the Safety Case

Engaging stakeholders on complex, and potentially contested, science

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Introduction

An effective process for engaging stakeholders on the science underpinning radioactive waste disposal will be essential for the successful implementation of geological disposal in the United Kingdom. Of particular importance are those stakeholders representing, and living in, volunteer communities. There have been two major shifts over the last 10-20 years in society's engagement with science which are particularly relevant to the Radioactive Waste Management Directorate's (RWMD) stakeholder engagement:

- a shift to a more inclusive approach in which the public have more of a say about science and its uses;
- a shift to a more evidence-based approach to societal decision making.

Significant challenges to effective communication and confidence building in geological disposal arise from:

- the complexities and uncertainties inherent in the relevant science;
- the sensitivities and "high stakes" (locally and nationally) associated with a disposal facility;
- the expectation that there will continue to be vocal stakeholders who are fundamentally opposed to geological disposal of radioactive wastes who will focus on any remaining uncertainties as just cause for their position.

This abstract summarises the findings of a project to evaluate approaches to engaging with stakeholders on the science underpinning sensitive decisions in sectors other than radioactive waste disposal and to identify elements of good practice which may help RWMD in taking forward the implementation of a geological disposal facility for the United Kingdom's radioactive wastes (Holmes, 2011). Six elements of good practice are listed and discussed below:

- using science appropriately;
- building trust;
- honest brokers;
- stakeholders as "scientists";

- communicating about uncertainty;
- protected spaces;
- using science appropriately.

There is a need to establish a framework to enable the appropriate use of science to inform decision making. An inclusive, evidence-based approach should draw on, and respect, a diversity of viewpoints from the scientific and lay communities, weighing them in a balanced and systematic appraisal of what is known, with what levels of confidence, and identifying where uncertainties remain which need to be addressed in order to achieve sufficient confidence to make a sound decision. It should identify where there is consensus, but also where differences of opinion remain and the reasons for those differences. A process should be established in which participants can challenge each other's views and evidence, but share a common goal of maximising the confidence that can be achieved in the knowledge and understanding that informs the decision.

Such a framework and process may be contrasted with more political models of decision making where decisions result from conflict, bargaining and coalition-forming among participants who each seek to protect or advance their particular interests. In such processes, scientific evidence is typically used as ammunition in a conflict, and called upon selectively by one or more parties to support their particular values and aims

Being selective of evidence has been prevalent in many recent public debates, for example those on climate change and genetic modification (Beddington, 2011).

Building trust

In the absence of the time, or the inclination, to engage personally with the science, many stakeholders' views are influenced strongly by whether they trust or mistrust the processes, institutions and individuals involved in a particular decision. Trust is hard won, but easily lost. Opinion surveys show that the UK public find it difficult to trust in business and government scientists (Ipsos MORI, 2011).

Various factors help to build trust in the processes whereby science informs decision making:

- The process should be open and transparent, with clarity about how the dialogue will inform the decision.
- Stakeholders who wish to be involved are enabled to do so, and can see that due account has been taken of their views.
- Realistic expectations of influence are established.
- Participants should have time to think issues through, and to become well-informed through reliable and balanced resources.
- Explanations are provided of how eventual decisions rest on the evidence.

It is also important to build trust at the interpersonal level between the staff and stakeholders who are involved. Good relationships, built over time, can be influential in influencing attitudes to the science: many stakeholders are more likely to evaluate the person than the science.

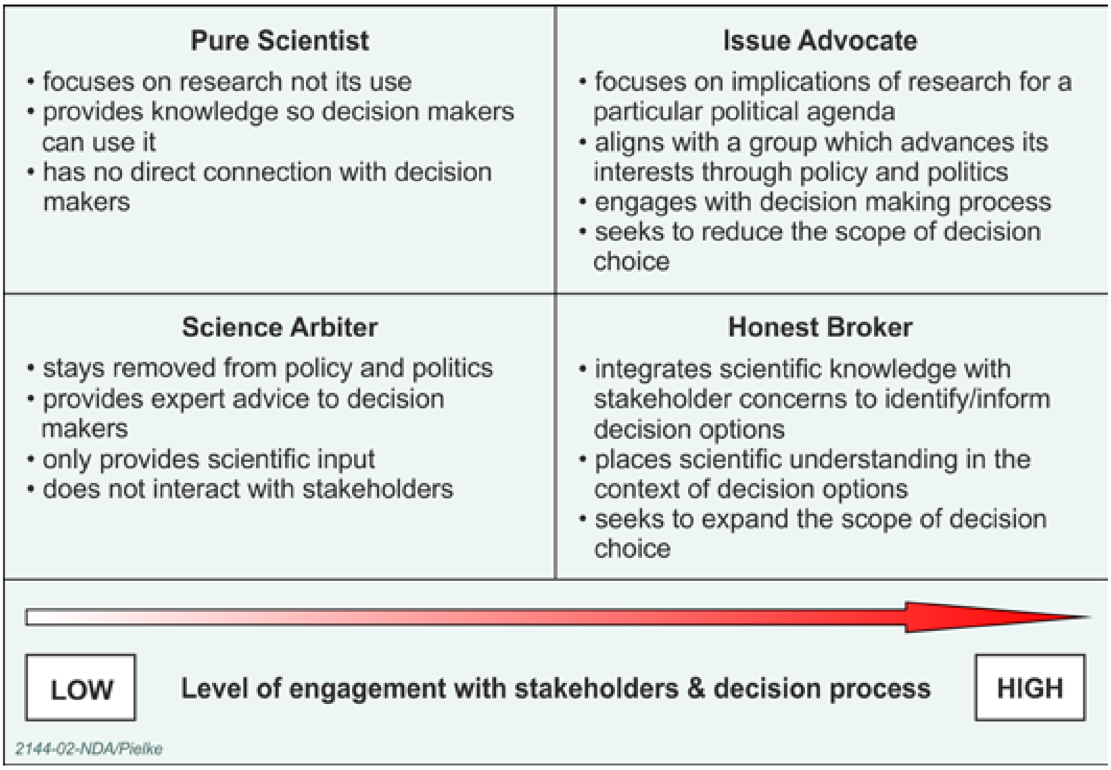
A research project on "deliberating the environment" examined how members of the public reacted to scientists in dialogues about environmental issues:

Their most positive reactions were to those two scientists who made most effort to find and develop common interests. Both of these scientists had considerable experience of public engagement but they also identified with their exchange partners. They were willing and *able to draw effectively on their own lives as "non-scientists" (e.g. as parents, partners, working class people, concerned citizens, confused "green" consumers) to make a connection with their exchange partner and promote conversation.* (ESRC, 2010)

Honest brokers

Honest brokers (Pielke, 2007), sometimes known as translators or knowledge brokers, can play an important role in facilitating interactions between science and decision making. Figure 1 illustrates how this role differs from other potential roles in scientists informing decision making.

Figure 1: Modes of scientist engagement with stakeholders and decision-making process



Source: Adapted from Pielke (2007).

A key motivation to establish and involve knowledge/honest brokers derives from the mistrust that stakeholders may retain of scientists employed by organisations who are proponents of the development on which a decision is to be made. Honest brokers need to be accepted by all parties as being impartial and authoritative, and must have a range of distinctive technical and communication skills in order to be effective (Bielak, *et al.*, 2009).

One example of honest brokers in action is provided by the establishment of an international panel of experts to review the issue of eutrophication in the Baltic Sea in 2005 (Holmes and Savgard, 2008). This issue was controversial, both in the sense of competing stakeholder interests across countries bordering the Baltic, and disagreements between scientists. Important factors in the success of the panel in helping to resolve controversies were the transparent approach to selecting panel members, and the freedom given to the panel in framing the questions and writing its report.

Stakeholders as "scientists"

Involving stakeholders in the generation and interpretation of scientific knowledge can be helpful in building trust through engendering ownership. Stakeholders may appropriately be involved in some, or all, of the following stages of the research process:

- informing the formulation of the research questions, ensuring that different framings of the issues are reflected, and consequently that research outputs address stakeholder concerns;
- undertaking the research, particularly where local and lay knowledge can improve the quality of the research;
- interpreting and communicating research so that it resonates with stakeholder concerns.

These considerations have led, for example, to the appointment of lay members to government advisory committees to help make expert advice more legitimate and encourage greater public confidence in the decisions of government.

For example, the Fisheries Science Partnership established by Defra in 2003 (and emulated in several European countries) involved fishermen in the co-commissioning of research on issues directly relating to catch quotas and regulations, issues of direct and substantial consequence for the fishermen (Holmes and Lock, 2008, 2010). This scheme has been very successful in building relationships and understanding between the fishing and science communities, and in securing ownership of the arising knowledge by the fishermen.

Successful stakeholder engagement in the research process requires significant investments of time and resource, and it is important to make adequate provision in research timetables and budgets.

Communicating about uncertainty

The science relating to issues of public significance always contains some degree of uncertainty. It is important that all stakeholders respect and accept this, as it:

- does not reflect a lack of quality or rigour in the science;
- does not mean that science loses its value in informing decisions.

Effective communication of uncertainties and their significance is rarely done well. Uncertainties and their negative consequences are often exaggerated by the media and by actors intent on a particular outcome.

The Intergovernmental Panel on Climate Change (IPCC) has devoted much effort to considering how confidence levels/uncertainties can be effectively communicated (IPCC, 2007). In the UK, the Marine Climate Change Impacts Partnership (MCCIP) has adopted a simplified approach in which confidence in annual statements about specific impacts of climate change is categorised as high, medium or low according to the amount of evidence that is available and the level of scientific agreement/consensus (MCCIP, 2011). This approach has proved popular with policy makers and stakeholders, and is now being extended by the Living with Environmental Change Partnership (www.lwec.org.uk) to annual statements ("report cards") for land-based impacts of climate change.

Protected spaces

In order to progress and facilitate decision making, spaces need to be created in which productive dialogue can take place between scientists and stakeholders. Issues and questions can be explored in a process of honest enquiry, without fear that views expressed will be used externally.

The Council for Science and Technology (2005) recommended that such spaces should be created which:

> *…provide a forum for reflective, considered and informed discussion between people with a range of views and values. Structured conversations between experts, non-experts and policy-makers can permit all to re-evaluate their perspectives and assumptions in the light of those of others, evolve their thinking, and explore areas of mutual and convergent understanding.*

For example, such spaces have proved effective in enabling productive dialogue between fishermen and marine scientists on the science underpinning catch quotas (Holmes and Lock, 2008, 2010).

Bennett (2002) points to the need to exclude the media from such protected spaces, "The presence of a journalist can jeopardise discussions. Stakeholders, thinking they may be quoted in tomorrow's newspaper, may not voice their true concerns."

Summary

Six elements of relevant good practice have been identified:

- *Using science appropriately:* Taking an inclusive, evidence-based approach in which collaborative inquiry takes a holistic, weight-of-evidence view of the science rather than focusing on items of evidence in isolation, using them to prove or disprove a particular point of view.
- *Building trust:* Ensuring that processes of engagement engender trust, and that trust is built at an interpersonal level between those involved in the process.
- *Honest brokers:* Can play an important role, mediating between the people and organisations involved, and interpreting the science and its significance for decisions.
- *Stakeholders as "scientists":* Involving stakeholders in the generation and interpretation of scientific knowledge promotes ownership and helps ensure that it is socially robust.
- *Communicating about uncertainty:* Establishing uncertainty as an inherent feature of science, and discussing uncertainties in a way which is helpful to stakeholders while remaining true to the science.
- *Protected spaces:* Creating spaces in which productive dialogue can take place between scientists and stakeholders, where issues and questions can be explored in a process of honest enquiry.

For RWMD the scientific uncertainties inherent in evaluating the performance of a geological disposal facility into the far future present a challenge of engaging productively with stakeholders on the science. The stakeholders who are fundamentally opposed to disposal will focus on remaining uncertainties as just cause for their position. Whatever the process of stakeholder engagement on science that is eventually developed, it will be important to build in evaluation and learning, together with the flexibility to adjust the process as experience is gained. Its development should also keep one eye on the planning and legal framework to ensure that synergies, rather than pitfalls, are built in.

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How Swedish communities organised themselves in reviewing a safety case

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In 2009 the Swedish nuclear waste management company, SKB, choose Forsmark in the municipality of Östhammar as the best place for a repository for spent nuclear fuel. In 2011, as a natural follow-up, SKB presented applications to two Swedish authorities, the Land and Environmental Court and the Swedish Radiation Safety Authority. The municipality of Östhammar has together with the municipality of Oskarshamn been an active part in the process since 1994, with different local organisations financed by the Swedish nuclear fund.

Three leading themes form the basis for our participation – voluntarism, complete openness of plans and results and participation with the possibility to influence.

Site investigations for a repository started in 2002 and were finished when SKB selected Östhammar municipality in the middle of 2009. To follow and scrutinise both site investigations as well as the applications, the organisation within the municipality has changed over time. As the site selection process got underway, the municipality extended its organisation to three committees (EIA, long-term safety and consultation). The committees have respective objectives: reviewing the health and environmental impact, reviewing the long-term safety and communication about the work that is going on within and around the municipality. These are primarily political committees to which a unit of civil servants is attached.

The main goal for the organisation is to build up knowledge skills and prepare both the existing as well as the future municipality council for the decision of whether or not the municipality of Östhammar will accept a final repository for spent nuclear fuel in our municipality.

The absolutely most important issue for the municipality is long term-safety and as the process has progressed the municipality has made several statements to the authorities.

Challenges of communicating safety case results to different audiences

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Introduction

Nowadays, nuclear politics and decision making are often oriented at procedures which are linked to precautionary concepts and which reflect forms of "knowledge politics" (Böschen, 2010). These precautionary concepts in most cases focus on robust societal decisions, which incorporate the principles of sustainability as a topic of public debate (Grunwald and Rösch 2012; esp. p. 10 and p. 12). The issue of high-level nuclear waste is under debate and confronted with public discourse, which integrates not only the knowledge of different stakeholders, but also accept certain forms of "Nichtwissen" ("non-knowledge").

Interdisciplinary research has to observe these normative trends and also has to "contextualise" these questions before interpreting its research results for giving answers with practical relevance, especially in communication with different social actors. Issues which are brought up in this field of nuclear waste management and their social context have to be analysed in two dimensions: *i)* the dimension of professionalism and expertise; *ii)* the dimension of managing controversial debates ("knowledge politics") and the preparation and implementing of robust decisions mostly by responsible governmental organisations. In this context on the one hand complex aspects of safety have to be communicated in their internal scientific logic and structure. On the other hand the different functional systems and collective actors of highly differentiated modern societies are engaged in controversial debates on advanced technologies like nuclear energy and technologies for waste disposal over long-lasting time periods.

Most safety and construction issues for final disposal of high-level waste, but also of waste management in general, are debated within professional "communities" of scientists and experts. But if their technological artefacts and their conceptual planning become issues of controversial and political debates in spheres which are outside the closed circle of high-level professionals and party politicians (who are in the end responsible for safety regulations, licensing and decision making), and also collective actors from civil society begin to discuss the side-effects of technological decisions like an underground repository, power relations and knowledge politics become more and more important (Straßheim, 2012; Stehr 2004).

The safety case discussion and its inherent impacts on public debate is an instructive example for the challenges which nuclear waste management has to face, if the public debate can be classified as a controversy which has to overcome societal cleavages between different groups of experts and civil society – cleavages which are articulated by modern mass media, civil society organisations and professionals with pro-nuclear as well as with sceptical or offensively anti-nuclear positions. Under these conditions, communication between very different types of actors (experts and others) is essential, but accompanied by considerable challenges. Safety is an issue about which all the relevant actors claim that it is of utmost importance, but at the same time it is an area with a high potential for communicative challenges.

Description of one typical challenge concerning the interpretation and communication of safety case results

Safety assessment stands at the core of a safety case (OECD/NEA, 2012, 2013; IAEA, 2012). It addresses safety during the operational and closure phases as well as safety after repository closure. Diverging views and experiences exist about whether operational or post-closure safety is of more interest and concern for those stakeholders who are not directly involved in the application and licensing process (i.e. concerned laypersons, established interest groups, protest groups and NGO and also intermediary actors like mass media and the new social media which organise the interested general public in society). Other issues such as environmental, economic or infrastructural impact might also be important. Beside the consensus that safety is always important, it is obvious that prominent stakeholders have different perceptions of safety problems. But their "framing" of the problems and the preferences and priorities for the problem solutions still need to be addressed by systematic case studies and empirical research [for the concept of framing see Benford and Snow (2000)].

In any case, implementers have to demonstrate and authorities have to judge compliance for the issues of concern, including operational and post-closure safety. The implementer has to mobilise support for this concept in a way that the public, especially in the vicinity of the facility, recognises the standards and methodologies applied within the professional safety case and assessment and at least to convince this local public to tolerate this type of assessment within a stepwise approach, where different modes of risk assessment and decision making are integrated. This all happens under unpleasing conditions, which influence the societal debate about safety in a serious way. Their main characteristics are: *i)* ongoing processes of "knowledge pluralisation"; *ii)* different types of sub-rationalities favoured by certain collective actors; *iii)* the necessity to decide now under conditions of an undeniable presence of different degrees and types of uncertainty and non-knowledge (or better, a systematic lack of specific, but relevant knowledge). These conditions describe the setting and in this sense important context variables for the impact of safety case concepts in a broader perspective. As safety case assessments are planned as instruments within iterative processes of decision-making, the three conditio mentioned above show that the instrumental logic within processes of planning and societal discourse challenge each other in a specific way, which is discussed here.

The safety case as a conceptual chance

For both operational and post-closure safety, safety assessments operate with scenarios (i.e. systematically derived but still postulated events and resulting future evolutions of the system), with their likelihood of occurrence and associated consequences. The latter is mostly indicated by estimated entities such as annual individual effective dose, annual risk, collective dose, etc. ("indicators"). Judging compliance means, amongst other things, comparing calculated or estimated indicator values with yardsticks prescribed in regulations (OECD/NEA, 2007, 2012b; ICRP, 2013). However, a safety case, being "a formal compilation of evidence, analyses and arguments that quantify and substantiate a claim that the repository will be safe" (OECD/NEA, 2013), contains quantitative as well as qualitative elements. Generating the named indicators means structuring, condensing and reducing a wealth of information and evidence supporting (or otherwise) the safety claim. Therefore, judging compliance cannot be reduced to a check whether or not indicators meet numerical criteria – the whole evidence as well as the methodology by which safety claims are derived from this evidence by means of the "safety case" tool is at stake.

Assessing operational safety for above-ground nuclear facilities is an established business. It is less established to achieve and assess operational safety for an underground nuclear facility, i.e. a repository. Specific challenges arise due to the necessity to address mining safety and nuclear safety at the same time. This paper, however, will focus on post-closure safety. When assessing post-closure safety uncertainties are higher and, consequently, calculated entities such as effective annual doses have a different meaning than usual (i.e. in classical radiation protection):

> *However, ICRP Publication 103 also warns that effective dose loses its direct connection to health detriment for doses in the future after a time span of a few generations, given the evolution of society, human habits and characteristics. Furthermore, in the distant future, the geosphere and the engineered system and, even more so, the biosphere will evolve in a less predictable way. The scientific basis for assessments of detriment to health at very long times into the future therefore becomes uncertain and the strict application of numerical criteria may then be inappropriate. In the very long term the dose and risk criteria are to be used for the sake of comparison of options rather than as means of assessing health detriment.* (ICRP, 2013)

In post-closure safety assessments, indicators like the ones mentioned above (so-called "safety indicators") but also others can be and are being calculated; in a recent review (OECD/NEA, 2012b), the following types of "complementary" (to dose or risk) indicators are distinguished:

- concentration and content-related indicators, that provide information on the radionuclide inventory and its distribution within compartments of the repository and the environment (e.g. total radioactivity content of the waste form or radiotoxicity concentration in groundwater);
- flux-related indicators that provide information on the transport of radionuclides between compartments of the repository and their release to the accessible environment (e.g. radioactivity flux from the engineered barriers to the geosphere or total integrated radiotoxicity flux from the geosphere to the biosphere over time);
- status of barrier-related indicators that provide information on the functioning and containment capability of the barriers in the repository system (e.g. container lifetime or buffer swelling pressure).

This is perhaps a more helpful categorisation than the conventional one (safety indicators, performance indicators, safety function indicators). The categorisation can then further be refined by asking about the location or system component it is related to, its purpose, etc. As examples illustrating the wide variety of indicators might serve (OECD/NEA, 2012a):

- container lifetime (status of barriers related), which might aid design optimisation;
- stress state in the confining rock zone (status of barriers related), which allows component-related performance statements and aids system understanding and design optimisation;
- groundwater age (status of barriers related), which allows performance and safety statements and aids communication;
- activity fluxes from waste container, concrete buffer, gallery, clay host rock, respectively (flux-related), which allows for performance statements about single system components and supports system understanding;

- radiotoxicity concentration in biosphere water (concentration- and content-related), which allows safety statements;
- extent of the potentially contaminated zone in the biosphere and the part of the geosphere located outside the disposal system (concentration- and content-related), which allows safety statements and might aid communication.

The review (OECD/NEA, 2012b) shows that implementing organisations usually have a clear view and strategy about how to use indicators *internally* and in their reports. Details of such strategies and terminologies vary considerably, though.

However, it is much less clear which indicators are of which use for which audience (authorities, interested public, scientific community, concerned laypersons,...) and which meaning any yardsticks or criteria for such indicators might have to them. By default, authorities will focus on those indicators for which regulations expect compliance (safety indicators such as annual individual effective dose or annual risk). However, developing regulations might also be related to concerns of media, the wider public, etc. An example illustrating how differently the indicators “risk”, “individual dose” and “collective dose” are perceived by different actors and audiences is provided in the following.

In 2010, the German Federal Ministry of the Environment, Nature Conservation and Reactor Safety (BMU) published its safety requirements for the disposal of heat-generating radioactive waste (BMU, 2010). In a preliminary draft (BMU, 2008), amongst other criteria two options for demonstrating radiological safety in the long term were offered between which the implementer was allowed to choose:

i) *The “conventional” or “traditional” approach:* Based on estimating contaminant release and migration to the biosphere, it should be shown that the additional lifetime risk of an individual to suffer a severe health effect caused by the facility will not exceed 10^{-4} (likely scenario) or 10^{-3} (less likely scenario), dependent on the likelihood of the scenario leading to this risk. In other words: the values address *conditional* risks (condition: the scenario under question will occur, its likelihood of occurrence is *not* aggregated into the value). Note that such conditional risk values are equivalent to effective dose values. According to BMU (2008) they translate, using the ICRP 103 risk coefficient of 0.057 (ICRP, 2007), to effective lifetime doses of 1.8 mSv or 18 mSv, respectively.

ii) *The “innovative” approach:* An indicator was introduced addressing the radionuclides released from the so-called confining rock zone. The confining rock zone (in the English version of the draft: “isolating rock zone”) is a concept which strives to achieve confinement by geologic/geotechnical barriers, the performance of which over the time frame of concern (up to 1 million years) can be forecasted with much more certainty than the evolution of overlying strata, hydrogeology or the biosphere. Thus, BMU’s safety requirements allowed for safety demonstration supported by an indicator related to this zone, by such means avoiding less reliable modelling of the hydrogeology and biosphere. The indicator was defined as annual effective dose to be calculated under the assumption that contaminated water leaving the confining rock zone (at some 100 metres depth) would directly go into a well which would provide for the whole water consumption of the individual under consideration. The yardstick was defined as 0.1 mSv per year.

In the ensuing discussion, this attempt to reduce uncertainties [approach ii)] was hardly ever mentioned, as the reasons for this shift in argumentation seemed obscure to most of the actors. However, severe criticism was expressed concerning the risk values to be applied for approach i). As a German daily newspaper has put it:

By this, many additional fatalities by cancer caused by a repository are possible, since the radioactivity released from underground might spread over large areas with thousands of inhabitants. Once initiated, the release might span over tens of thousands of years and *would affect many future generations. After all, highly radioactive waste will remain hazardous for one million years.* (TAZ, 2009)

For radioactive waste specialists, indicators such as risk are embedded in a technical context which is often (as in the BMU draft) not explicitly stated. Selected aspects of this context are discussed below:

1) The requirement to meet numerical (e.g. risk) criteria is seen by specialists as just one amongst multiple lines of reasoning to be made in a safety case. The BMU draft spent only one or two of its 25 pages on the requirement mentioned above. Still, discussions with decision makers, media or other concerned laypersons tend to focus on numerical criteria. This was also the case when the draft safety requirements (BMU, 2008) were discussed; the discussion was mostly restricted to three issues, two of which were directly related to the risk criterion. It was questioned whether or not the use of a risk criterion in itself was appropriate and whether or not the choice of the numerical values was appropriate. (The third issue, and the only one not related to numerical criteria, concerned a retrievability requirement imposed in the draft.)

2) Perhaps this misunderstanding on the role and importance of numerical criteria is caused by the perception that they are the only "hard" or "verifiable" safety requirements. Experts from the safety case community have a different view. Their calculation (or rather estimation) of risk (or dose) values in the far future has to rely on assumptions concerning the future evolution and states of the hydrological system as well as on pathways in the biosphere, food chains, exposure modes, nutrition habits, etc. Especially the latter can hardly be predicted over time frames exceeding a couple of years or, at the most, decades. Models for estimating dose or risk are based on "stylised assumptions" concerning these issues, against the choice of which they are often quite sensitive. Therefore, "...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." (ICRP, 2007) BMU's approach ii) described above aimed at avoiding the uncertainties associated with hydrogeological and biosphere models reaching into the far future. But while this approach with its merits and problems is up to today extensively discussed by German experts, other audiences hardly take notice of the underlying general idea.

3) The stylised assumptions mentioned above aim at conservative estimates of radiological consequences. Therefore, they will often include variants in which as many as possible of the contaminants released to the environment are assumed to contribute to the exposure of only a few, but "highly" exposed, individuals (e.g. self-sustained farmer models). The dose or risk yardsticks will then be compared to these exposure estimates. If they are met, it is likely (but admittedly not guaranteed) that other models in which the contaminants are more diluted and hypothetically expose "thousands of inhabitants" (TAZ, 2009) would lead to much lower risk estimates.

4) The dose values equivalent to the risk target (effective lifetime doses of 1.8 mSv or 18 mSv) are by far lower (1.8 mSv) or in the order of magnitude of (18 mSv) the variability of exposition from natural background radiation. Epidemic evidence for deriving a dose-risk relationship for such low dose values is poor, by some it is even questioned whether such a relationship exists. Although – as recommended by ICRP – a linear dose-risk relationship is assumed as a basis for rule making, many specialists will probably not translate such dose values into a number of *actual* cancer cases but simply qualify them as "tolerable". Such an "experts' attitude" is also supported by an awareness that other (radiological and non-radiological) risks people are exposed to in daily life are often higher (and sometimes considerably higher). And if an expert does not think about *actual* cancer cases, he/she is not likely to change this attitude just because the timespan of concern is long.

Consequently, the attitude of a radioactive waste specialist and his/her perception of the BMU risk criterion will be considerably different from the one expressed in TAZ (2009). Specialists might even believe that risk indicators are better suited for communication with non-specialists than dose indicators, because they can be compared to (known) risks of daily life, e.g. of traffic accidents. The newspaper article quoted above is an indication that this perception might be awfully wrong, and that much more care and skill is needed when communicating safety assessment results to non-specialists. One might even speculate that a severe criticism as described above had never been voiced if the BMU draft had used the dose criteria equivalent to the risk values from the beginning.

Responding to discussions in political bodies, with stakeholders and with concerned laypersons, BMU revised its draft safety requirements. In the final version (BMU, 2010), the risk values mentioned under i) above were replaced by criteria for annual effective dose (10 mSv for likely and 0.1 mSv for less likely evolutions). In order to address concerns about contaminants affecting numerous people and despite the concerns expressed by many specialists (e.g. ICRP) about the use (or rather uselessness) of collective dose in long-term safety assessment, BMU decided to use an indicator based on the concept of annual collective dose for limiting the releases from the confining rock zone [i.e. for approach ii) from above]. The criteria are 0.1 person-mSv per year for likely scenarios and 1 person-mSv per year for less likely potential evolutions (scenarios).

This illustrates how differently indicators can be perceived by different audiences: While the specialists were focused on avoiding modelling uncertainties by introducing the alternative criterion ii) and had no concern at all about the sufficiency of the “traditional” criterion named under i), other audiences were totally “ignorant” of this aspect and instead emphasised their misinterpretation of the meaning of the indicators (conditional risk or annual effective dose), especially for low dose values.

More questions can be asked about the relevance of different indicators.

Dose versus risk

It is often claimed that risk indicators are more suitable for communication compared to dose indicators since their values can be compared to other risks of daily life. On the other hand, risk values are entities which aggregate many different types of information: dose (by itself being an indicator), dose-risk relationship and likelihood of occurrence. Is that too much information, given that already dose is an entity with a high degree of aggregation (release, migration, radionuclide uptake/exposure modes, radiology) and is therefore hard to understand? Which is indeed “better” for which audience? Does the example from above indicate that, on the contrary, risk indicators should be used and communicated with the same care as dose indicators?

It should also be noted that the “daily use” of the risk concept in other businesses (e.g. about traffic accidents) implies that some people indeed will be killed, but society accepts this since it is a small percentage of the total number of people concerned (e.g. by travelling by car). This is a concept different from that of disposal; just one facility will be constructed, the number of people belonging to a potentially exposed group remains small (at least if one point in time is considered – cf. above) and calculated risk figures multiplied by this number of people might result in a figure much smaller than 1. This means that, provided that there is no risk dilution, most likely nobody is going to be killed (not to speak about the unclear dose-consequence relationship for small doses). What does this tell us about the risk indicator as a means of communication?

Values (or otherwise) of flux-related indicators

Do “flux-related” indicators in general convey the message intended? After all, disposal is about containment, not about release. Calculating releases might therefore be amenable to misunderstanding, especially since it is difficult to communicate that the calculated releases are "negligible". Wider audiences might either not know that there is such a thing as "negligible release", or they might disagree.

Confidence

More generally, the question arises about the confidence different audiences have in the safety assessment calculations. Such calculations do not forecast the future, rather, they are meant to demonstrate that uncertainties can be bound and containment will be achieved with a high degree of confidence. They are based on many assumptions (about scenarios, physico-chemical processes, etc.) which have to be supported by different lines of evidence, tested against alternative assumptions, etc. Usually, the modeller does not "believe" in the exact figures he/she produces, but nevertheless gains confidence ("Models are about insight, not about numbers.") The challenge is to organise the time-consuming substantial debate with the interested public, independent experts, responsible authorities and stakeholders about the tracing pickets of orientation, generated by every professional safety case. What is the minimum of messages to be translated to, and discussed with the general public?

Multiple lines of evidence in a safety case

The confidence issue explained above is one of the reasons for having created the modern concept of a safety case, in which assessment calculations form one of several lines of evidence. Which of these other lines of evidence (lab or field results, technical and natural analogues, *in situ* information e.g. about groundwater ages, engineering framework, quality assurance, verbal description of the safety concept,...) are helpful and understandable, which are more or less not?

Passive safety

Amongst specialists, the concept of passive safety (no reliance on active safety measures such as monitoring, surveillance, refurbishment) is considered as a strength of geologic disposal. Amongst non-specialists this is not necessarily the case – many of them have more confidence in active measures. Can something be done about this inconsistency? Is it helpful to deviate from the idea of "definitiveness", as is discussed in Switzerland?

Addressing uncertainties

Modern safety cases are about creating confidence by multiple lines of evidence. But they are also about systematically compiling and analysing uncertainties and open issues in order to derive strategies for addressing these issues. In fact, messages concerning the latter are the main drivers of a disposal programme. Apparently, uncertainties also have the potential for miscommunication. If they are not made visible, perceptions like "these folks are over-confident" or even "they want to betray us" might be the result. If they are too pronounced, perceptions like "decades of research, and these amateurs still know nothing" are possible (the authors have already experienced both). What can be done?

In summary, and more generally, the following three questions can be asked about any kind of information documented in a safety case:

1) Is it understandable, or, in other words, does it carry across the same information for different people?
2) Is it considered relevant?
3) Is the information credible for diverse audiences? To what extent does its credibility depend on the trust in those who generate it (mostly waste management organisations)?

Is there a chance for professional concepts?

Especially the international discussion gives some important hints for concepts of professional communication about safety case concepts and the results of modern safety cases, which integrate social problems of nuclear waste management in their design.

It is generally agreed that a safety case documentation should be structured hierarchically. The French Dossier 2005 (Andra, 2006) established a sophisticated example for doing so by establishing five levels of documentation:

1) a leaflet of four pages aimed at the general public, a brochure (38 pages) for concerned laypersons, a synthesis report (about 200 pages) for decision makers;
2) three synthesising and transversal reports (on architecture and management, phenomenology and safety assessment, each between 500 and 700 pages);
3) five "knowledge reports" on different issues (e.g. material sciences, geosciences, etc., each between 500 and 1 000 pages);
4) 72 technical documents (some dozen – some 100 pages);
5) several informal documents, articles in scientific journals, etc.

From Level 2 on downwards, the intended audience becomes increasingly specialised. For these "specialists", as well as for generalists trying to trace a certain information ("Where does this sorption value come from?") traceability is paramount, and the way of distributing the available information amongst the different documents follows scientific standards and standards of "public understanding of science". The most interesting issue for our problem of communicating safety case results is, however, Level 1: Which type of information has to be presented in which document and in which formats, all aimed at three different groups: interested public, concerned laypersons and decision makers? How has this information been presented? Furthermore, and going beyond issues related to written documentation, questions arise about opportunities and challenges related to other media (mass media, world wide web, social media). How can consistency with the written documentation and related messages be ensured?

Nevertheless, the safety case framework which allows presenting different types of evidence in different formats opens possibilities for communicating safety-relevant messages to different audiences. However, it has to be ensured that these messages are consistent with each other (there is only one safety case at a time) and they are perceived in a consistent way. The central aspect is that the results and arguments coming from safety case analyses have to be accompanied by more (rather than less) intensive communication with the interested public and stakeholders over time. In our perspective the safety case tool in general is an analytical instrument for foresight and has an integrative and central function within the licensing procedure. But on the other hand the instrument also has its limits. As with every case of foresight it is a professional assessment. It is based, *inter alia*, on science, engineering and modelling, and from interdisciplinary risk research we know that "numbers" give orientation, but real-time processes and experiences on actual sites will have their own logic once the facility is constructed. As so far no actual experience with high-level waste repositories has been made and decisions have to be taken now, all need a qualified dialogue about tools for gaining orientation and sufficient confidence (and by all its limits especially as it is foresight), the professional quality of an assessment of this type has to be debated and the limits of this tool have to be documented. If decision-making is blocked over time, this non-decision is also a decision with often extremely negative side effects. Lost resources (intellectual and financial), or, even more importantly, loss of safety and security of prolonged waste interim storage, can be such a type of side effect.

Lessons for the real-time experiment

- The public is a highly relevant sub-system in the processes of technological and societal innovation especially in cases like waste management. It is the third anchor beside polity and economy in the case of nuclear waste.
- Following the German sociologist Max Weber with his *trias* of polity, economy and culture we have to reflect, that the sense ("Sinnhaftigkeit") of collective action within these different systems influences the struggle for the correct solution for our specific type of waste as one type of high-problematic waste. As culture is on the one hand not only structured by one abstract and general binding rationality, but also by functionally differentiated sciences and their disciplines with their own systematic sub-rationalities, societal consensus and respect for different positions can be gained through dialogue. On the other hand forms of societal self-organisation and constructive debate stabilise culture and form the basis of society. All these forms of aggregation and organisation by science and social self-organisation offer a specific knowledge output. Mobilising local citizenship initiatives, pre-political national and/or sub-cultural networks or extra-parliamentary protest groups (or as an aggregated form social movements like the anti-nuke movement) over time became part of these cultural networks and by this with their own knowledge and expertise part of the complexity of the problem, which has to be managed by radioactive waste actors. Culture in current modern societies (as a fact) is highly differentiated, structured by strong cleavages and enforced by a wide range of value patterns and stabilised by more or less closed and often technology based forms of (expert and sometimes mass) communication.
- In this complex field of culture with its often more or less small integrating networks, four types of public are relevant to our safety case issues. These types of public have to be discussed separately: *i)* experts and scientists (with their forms of institutionalisation like universities and disciplinary associations); *ii)* laypersons; *iii)* political public sphere; *iv)* mass media.

  i) *Experts and scientists:* In the last four decades nuclear waste management became a more and more institutionalised field of expertise with strong links to the power industry. Some civil protection authorities (*Schutzbehörden*), such as federal offices for radiological protection and departments of environmental ministries, and their private partners, which were hired continuously as consultants and service providers, were strongly connected with this field of expertise, long dominated by engineers and natural scientists. Often like a closed shop they were developing engineering- and safety-related research and developed solutions for the nuclear waste problem with a wide range of different characteristics. Their research was and is highly elaborated and in most cases structured by a limited number of dominant mainstream positions. Especially in countries with stronger anti-nuclear opposition a sub-sector of counter experts has developed and been established over the decades and sometimes integrated in research and consultation. Often these counter experts founded small institutes with service units for environmental planning and consultation. Like the German Oeko-Institute (Freiburg/Darmstadt), they can be integrated in the national discussion about safety case development. Despite the differences with regard to access to resources the level of expertise and quality of consultation is very near or in many cases on the same level as the traditional, well-established communities of nuclear scientists.

  ii) *Laypersons:* It is not surprising that in most cases laypersons are far removed from these discourses, research and reflections; their level of knowledge of site-specific safety aspects and the progress of nuclear waste management in this field is not very developed, as for example Eurobarometer data show for most European countries (2005, 2010). The question is, which type of sorting indicators along the line "positive for public communication"/"challenging for public communication" can be more helpful for societal impact, i.e. showing the robust results of safety case studies. The question can only be answered by estimating in a rough way, as basic research and systematic empirical work on high-level standards is not available (OECD/NEA, 2012b, p. 70). Specific indicators like radiotoxicity concentration in groundwater or flux-related indicators seem to be more helpful in public communication, as the lay people to whom communication is aimed have an understanding of the context. Indicators, however, generate information by means of highly abstract numerical models relying on academic codes of communications; scepticism with this type of abstract information, where numbers are relative measures in complex systems, is difficult to overcome. Following the suggestions to offer results from safety case analysis in practical examples (for local contexts or impact significance) and in the context of two way communication makes sense (see the instructive example in OECD/NEA, 2012b, p. 55). On the other hand, from German experience it must be expected that there is a sometimes erratic "learning curve": Every communication in this sense generates new questions, which do not necessarily account for the messages learnt from the constraints that already were accepted when answering the previous question. Laypersons are in many cases wise and headstrong enough to ask for unexpected futures or expositions in certain socio-economic situations, which are only interesting in an abstract way. They do not do this for irrational reasons, and they often have the wisdom to detect aspects which are really challenging for even the best experts in fields such as nuclear waste (Wynne, 1996).

iii) *Political public:* The political public differs enormously from country to country. This depends on collective experiences with civil nuclear energy and also in some cases with military interests. In Germany, the highly polarising conflict in party politics and society over more than three decades and the "muddling through" strategy employed by the responsible governmental organisations (Hocke and Renn, 2009, p. 930) resulted in safety politics becoming a domain which follows a strategy of defence and/or scandal. The question of political majorities at the state or national level seem to constantly dominate public discourse, mostly run by communication channels of (new and old) media. Professional expertise using numerical tools for assessing safety by indicators for specific contexts (like waste disposal) are very likely to be important in spatial planning and licensing processes, as they fit to the general aims of planning and licensing on the basis of safety standards. These procedures of planning and licensing have to be embedded in "robust" political decisions. But interest aggregation and the development of a very small number of alternative safety concepts do not seem to be issues fitting to the discourse of political profiling and gaining advantages in election campaigns, which are preconditions for political consensus on basic assumptions necessary for preparing robust decisions.

iv) *Mass media* (old ones like quality press, television or new like blogs and other social media): Mass media are important for connecting individual members of a complex and differentiated society, especially today, with the high rate of "individualisation" and "pluralisation". But their forms of selection modes are structured along a small group of news factors. Following the discussion and robust results from communication research there is no chance or only a very limited chance to obtain a reflected description of safety case research and communicating safety case results in mass media. News factors like "conflict", "crime" and "scandal" steers the selection of mass media coverage more than the relevance of technological policy (Eilders, 2006). Under certain conditions mass media discourse offers the chance of strengthening convincing arguments (Böschen, 2010, p. 118), What is the conclusion of this analysis for safety case experts? One central message is that safety case experts have to be prepared for complex debates with very different groups of society. In these discussions their main duty will be to explain the complex safety case models as a professional assessment and method of foresight. This includes being prepared to explain the sense of single indicators and the limitations of complex models on one hand, as well as the undeniable advantage of this type of safety calculation and safety assessment on the other. Honesty, especially about (as yet) unresolved issues is strictly necessary, but has to go along with skilled communication about the reasons for the existence of such issues (despite decades of repository research), their significance e.g. for safety (or otherwise), and perspectives to resolve them in the future (or otherwise). The necessity of being prepared for ongoing processes and often repeating translations of scientific knowledge into other types of knowledge, which can be understood by well-informed engaged academics with an open interest and other social groups, over time, will be one central task during the coming years and decades. For research, it is an open question whether communication about safety case issues can widen the horizon of the respective national debate in countries developing nuclear repositories in a way that the two analytically challenging questions of, first, how to foresee evolutions of the disposal system and, second, how to make confidence in evaluation procedures become more important in siting and implementing than regional competition and dissent between political parties. Every collective decision taken by the current generation will be a decision accompanied by a number of uncertainties and will also be taken under conditions of uncertainty. The political culture must manage dissent on one hand, and recognise that conflict and dialogue between different groups and stakeholders is a central tool in civilised societies to find robust and legitimate solutions.

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Demonstrating safety: Lessons learnt by InSOTEC

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Introduction

InSOTEC is a three-year collaborative social sciences research project funded under the European Atomic Energy Community's 7th Framework Programme FP7/2007-2011, under grant agreement n°2699009.[1] The project aims to generate a better understanding of the complex interplay between the technical and the social in radioactive waste management (RWM) and, in particular, in the context of the design and implementation of geological disposal.

In doing so, InSOTEC wants to move beyond the social and technical division by treating RWM and geological disposal as "socio-technical" challenges and in following the relationship and describing the context, one can identify the dependency as a socio-technical combination.

InSOTEC focuses on situations and issues where the relationship between the technical and social components of geological disposal are still unstable, ambiguous or controversial, and where negotiations are taking place in terms of problem definitions and preferred solutions. Some concrete examples of socio-technical challenges are the question of siting and of introducing the notion of reversibility and retrievability or long-term repository monitoring into the concept of geological disposal. These examples show that the concept of geological disposal develops over time, not only because of evolutions in scientific knowledge, but also as a consequence of debates on how to implement this technology in the light of societal requirements.

During the first year of the project, various research activities in the national context of InSOTEC partner countries as well as on the European and international levels contributed to the identification of the main socio-technical challenges in geological disposal.

On this basis four topics were selected for in-depth analysis:

- reversibility and retrievability;
- demonstrating safety;
- siting;
- technology transfer;

1. InSOTEC partners are: the University of Antwerp (Belgium), the University of East Anglia (UK), Öko-Institut e.V. (Germany), Göteborg University (Sweden), CNRS – Ecole des Mines de Paris (France), MTA TK (Hungary), GMF (Spain), the University of Tampere (Finland), the University of Jyvaskylan (Finland), the University of Ljubljana (Slovenia), Charles University (Czech Republic), Merience Strategic Thinking (Spain), the University of Oslo (Norway). The project started in March 2011 and terminates February 2014. For more information see www.insotec.eu.

The aim of these analyses is to come to a better understanding of the relationships between social and technical challenges in geological disposal. It is being investigated how these relationships are becoming visible in various "social-technical combinations", depending for example on the actors involved and the issue at stake, see also Bergmans, *et al.* (2012).

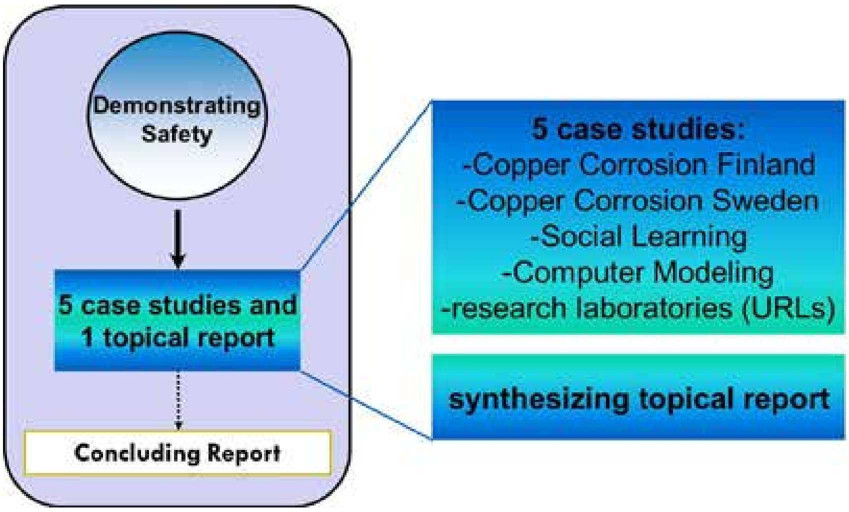
For the time being the research work on these topics is still ongoing. Case study reports and topical syntheses are to be finalised. An overall "concluding report" (see Figure 1) of the main findings is planned to be published at the end of this year.

This paper, therefore, presents work in progress and the findings on "demonstrating safety" as a socio-technical combinations are preliminary.

The InSOTEC approach for in-depth analysis of "demonstrating safety"

The structure of the research approach is visualised in Figure 1.

Figure 1: Structure of InSOTEC research activities on demonstrating safety



The case studies are performed by InSOTEC partners or small teams of partners from different countries. They cover a broad spectrum of safety-relevant issues, national experience and backgrounds.

The synthesising topical report is developed by Oeko-Institut with the support of the authors of the case studies. The synthesising approach and current findings of this work in progress are the focus of the following sections.

Synthesising the findings on "demonstrating safety" as a socio-technical combination is based on a combination of methodological steps. At the beginning an overall framework for the development of case studies was defined by a set of questions which have been developed on the methodological basis of science and technology studies (STS) (Bijker and Law, 1994):

- Which (technical) innovations or (institutional) changes ("socio-technical modifications") have you discovered/identified within your case? What was the starting point of the process (initial programme) and which key issues have changed by what kind of discourse?
- Which tools (or concepts) have facilitated the socio-technical modifications?

- How would you describe the robustness of the technical or institutional changes/ solutions? Which are remaining problems, contradictions and ambiguities?
- How did actors and actors-networks contribute to the socio-technical modifications? How is the role of actors and actors-networks influenced by these modifications?

These questions reflect that modifications in either the "social" (e.g. political preferences, public concerns, social values, national traditions, assumptions about future generations, decision-making processes) or the "technical" [e.g. the kind of waste stored, the disposal concept with regard to geological characteristics of the site or the properties of containers, (tools for) measurements] are one important indicator of socio-technical combinations. Furthermore we learn and hopefully better understand the technical change – as our starting point – by analysing the way a technical "programme" is confronted with social "anti-programmes" and the flexibility to integrate new challenges as discussed in Latour (1991).

These findings will underpin the assumption that a technical solution is always a response to a political situation, and as such forms a socio-technical combination that also avoids critical "hot situations" and an "overflow" of controversies as described by Callon and Law (1989). Another analytical concept deals with the effects of failed integration and the missing adaptation to new political concerns. This implies that the solution can become obsolete and leads to overflow and dissatisfied groups, who do not accept (anymore) the delegation of the issue to a given technical solution (Barthe, 2009). As a consequence, the possibility of a political reformulation is also given, questioning an earlier decided delegation to technical solutions (Kall and Sundquist, 2013).

The set of questions has in the following been further refined to reflect the specific needs of the demonstrating safety topic.

In order to further examine the broad spectrum of issues that is touched on in the context of demonstrating safety, the three "dimensions" which are covered by the guiding questions have been explicated by using examples from the case studies and from experience:

- Substantive dimension:
  - concepts and technologies which are planned to be used to isolate nuclear waste;
  - technologies and programmes which are planned to be used to check safety/ to check if the repository behaves as expected;
  - organisational/management needs and structures that assure the long-term availability of personnel and financial resources;
  - environmental and socio-economic impacts;
  - safety and environmental regulations.
- Procedural dimension:
  - development and update of safety and environmental regulations (process and actors);
  - development, definition and update of waste management policy and disposal concept (process and actors);
  - dealing with uncertainties in the context of evolving knowledge;
  - formal and informal parts of siting and licensing procedures and the interaction of actors;
  - strategic environmental assessment (SEA) and environmental impact assessment (EIA).

- Methodological dimension:
  - methodologies for safety analyses (modelling, scenario building, calculations, analogues, etc.);
  - the safety case as a method for presenting analyses and arguments as a basis for major decisions, a tool for communication,…;
  - analyses of environmental and socio-economic impacts;
  - knowledge management and social learning;
  - stepwise approach and iterative optimisation;
  - discourse and justification.

The three dimensions are closely interlinked. They are not intended to be used for strict categorisation of issues to one or another dimension, but rather to support the analytical understanding of how different factors can contribute to demonstrating safety.

The "safety case" may serve as an example that illustrates the close interrelation of the dimensions and the plurality of an issue: During the different phases of a disposal project (planning, siting, licensing, construction, etc.) the safety case provides a method for presenting analyses and arguments on the safety of the project in a structured way. It was thus presented under the methodological dimension in the examples above. However, by presenting a safety case at specific stages of the project it also touches the procedural dimension as it provides an important milestone with regard to the interaction with authorities and to communications with the public. Last but not least it can be expected that the comprehensive analyses that are performed in a safety case also influence the way that technical or organisational matters are planned to be solved.

"Demonstrating safety" – preliminary results of synthesis and analysis

First analyses of the five draft case studies, listed in Figure 1, have been performed using the scheme of the three dimensions in order to enhance the understanding of the respective cases. They revealed that each case covers all three dimensions with the focus being different between the cases.

As outlined for example by the case of underground research laboratories (URL), it fits well within the context of demonstrating safety and making safety arguments, but a URL in itself – particularly a generic one – is not a "demonstrator" in a specific safety case. Therefore, it was not possible for them to address all issues at the same scale.

URL – similar to other cases – offer a basis for safety arguments. They provide, in a sense, the infrastructure and the conditions for scientists to elaborate arguments about safety and security. They are a place for research and development concerning technical solutions and a place of experimentation. On the other hand, URL have become increasingly concerned with communication with various audiences and thus, they also have shifted more and more to a mixed platform for RD&D and public communication. In this sense, the social and the technical have been combined via the diversity in an URL's missions and roles.

Due to the example of URL and the computer modelling case, it became obvious that one cannot really demonstrate safety per se. Caused by the long-term dimension one can only try to demonstrate that one knows about the factors supporting safety on the one hand and about expected risks on the other hand, that should be taken into account by designing a concept.

In a second analytical step – after cases are typed – a preliminary first collection of indicators of safety has been compiled and characterised: e.g. awareness rising factors have been identified and prioritised as indicators of change and socio-technical combination.

One major issue that has been observed across all case studies is an (institutional) change in communication and communicative structures. Secondly, gaining new actors and the building up of actor networks is seen an important aspect of integrative socio-technical processes.

Acknowledgements

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Safety, safety case and society – Lessons from the experience of the Forum on Stakeholder Confidence and other NEA initiatives

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Setting the scene

Governance of radioactive waste

A vast amount of literature on radioactive waste management (RWM) and its governance is available on the webpage of the Radioactive Waste Management Committee[1] of the OECD Nuclear Energy Agency (NEA), in particular on the pages of the Forum on Stakeholder Confidence (FSC),[2] the Reversibility & Retrievability (R&R) Project[3] and the Project on Records, Knowledge and Memory (RK&M) Preservation across Generations.[4] The FSC literature alone likely represents the largest collection of literature on RWM governance presently available on any single site.

On the safety case

The safety case developed for any deep geological repository project deals with technical safety. A license is to be granted based on the repository being, after closure, safe "by itself", i.e. without the need to watch it, independent of the existence of the implementer, regulator and others.

The main legal requirement of the safety case is that it needs to show convincingly that the technical regulatory criteria are met. The latter are both qualitative and quantitative. Qualitative criteria are technical, but not in a strong sense, e.g. one requirement may simply be the use of "sound technical and managerial principles". The safety case also needs to argue robustness upon human intrusion. The human intrusion analyses, however, are only used to make a qualitative judgement on the robustness of the system. The international guidance suggests that their results need not be tested, by the authorities, for compliance against a numerical yardstick (ICRP, 2013).

The technical regulator will have an important role in decision making, but others aside from the technical regulator will also play a decision-making role in the development of a repository project and with regard to its safety. For instance, the technical regulator is largely removed from the initial choice of site.

1. www.oecd-nea.org/rwm.
2. www.oecd-nea.org/rwm/fsc.
3. www.oecd-nea.org/rwm/rr.
4. www.oecd-nea.org/rwm/rkm.

The meaning(s) and components of "safety"

What is safety?

Safety, in the technical sense, means *no significant threat to people's health and the environment*. In fact, it means *adequate* protection given present-day values and circumstances.

Safety in the broader, civil-society sense means: *feeling safe*, *free from danger*. Components of *feeling safe* are: knowing that a trustworthy individual, group or institution is in charge; being familiar with the issue; being able to exercise oversight/control.

Trust – familiarity – control

The understanding that safety is also about trust, familiarity and control was an important finding of the FSC (OECD/NEA, 2004, 2013a). The experience of other NEA initiatives, such as the RK&M and the R&R projects, underscores the importance of these components of safety as well (see below, *Related findings from RK&M and R&R projects*).

How different is it in the "technical" field?

The safety case is not a reproducible scientific experiment. Two separate teams may come up with different results, albeit these results may not be seen as significant by the technical specialists.

Technical specialists use a combination of both cogent technical arguments and also a set of more subjective elements to develop and/or review a safety case. Namely, to mature their confidence, they rely on their familiarity with the project and the work that was performed; their knowledge of the history of the project, of the trustworthiness of the implementer and of the rigour of the regulatory system, for instance.

Institutional key actors, such as regulators, developers and scientific committees, also have or have had the opportunity to exercise control, which gives/gave them the possibility to direct the project in the direction they think/thought best.

It can be thus affirmed that trust, familiarity and control also play an important role for the specialists when judging technical safety.

Related findings from RK&M and R&R projects

Reversibility & retrievability (R&R)

Reversibility and retrievability are typically presented as a societal demand and as assurance not to commit *irreversibly* to one path and to show, instead, adaptability and flexibility. Demands for reversibility and retrievability underscore the desire for forms of continued responsibility and control.

Monitoring

Monitoring, after closure, has been viewed in technical circles as a practically useless endeavour. During the decades-long period in which a repository goes through different phases up to final closure, a consensus may develop that monitoring after closure may not be needed. For the moment, however, communities do ask for monitoring to take place and also, in some cases, for a role in monitoring both during and after closure (OECD/NEA, 2013b).

The FSC provides examples of the importance of this issue to involved local stakeholders: Hungary (control of waste packages); France (independent environmental and health monitoring); Belgium and Nye County, Nevada (proposed sets of indicators). A study on monitoring contributed by the FSC to the RK&M project provides further evidence of the importance of monitoring to local communities (OECD/NEA, 2013b).

Records, knowledge and memory (RK&M)

RK&M preservation is emerging as an item of interest and concern in local communities as well (OECD/NEA, 2013b). There is growing attention to these matters by local communities. FSC examples include the workshops in Meuse/Haute Marne (France) (OECD/NEA, 2012) and in Östhammar (Sweden) (OECD/NEA, 2010).

Oversight

The modern concept of oversight as exposed in ICRP Publication 122 (2013) militates in favour of monitoring and RK&M preservation; oversight is characterised by the ICRP as a needed, continuing societal endeavour. Initial findings by the RK&M project on oversight and monitoring were presented at the MoDeRN conference in spring 2013 (Pescatore, 2013).

Stakeholders, technical specialists and safety

Safety is, in the first place, the responsibility of the implementer.

Safety existed before the technical regulators were created roughly 40 years ago. The contribution of the technical regulators – whose agencies initially were not as independent of ministries and governments as they are today – is to increase the chances of a safe project by reducing bias on the part of the implementer and by making decisions more transparent.

As democracies have developed further over the past 40 years, the public has shown increasing interest in having a say, especially on controversial projects that may affect their lives. One of the reasons for the public interest is the failure of national systems to prevent major crises, such as mad cow disease, the recent world-wide financial crisis, Chernobyl, etc. Today, the public and other specialised groups, e.g. national committees, are often given specific powers in specific circumstances and are part of what may be seen as a broader regulatory system set up by society, which also includes the technical regulatory bodies.

The questioning of the institutional players – their studies, their behaviour, etc. – by the public and other bodies, be they implementers, regulators, political leaders, etc., constitutes another contribution to making decisions more transparent and to reducing bias. Again, this increases the chances of a safe project.

The way safety is delivered has changed over time, reflecting the progress of democracy. Safety nowadays is brought about by a system of actors that comprises the implementer, the technical regulators, specialist groups in various advisory roles and the public, even if the legal responsibilities are different from one actor to the other.

FSC specific questions on safety and the safety case

The FSC Annotated Glossary of 2013 states that the safety case must provide convincing evidence of the level of understanding and control for the public, and must ensure that no question remains without a well-founded answer.

According to the FSC, open questions of interest include (OECD/NEA, 2013a):

- What is long-term passive safety?
- How are technical and subjective elements brought together?
- How do we explain passive safety to the lay public?
- What is the link between safety and the several degrees (or gradual removals) of control?

Integrating civil society into the safety case process

From the FSC "Stepwise Decision Making" document of 2004 (OECD/NEA)

Within the RWM community it is now broadly agreed that any decision making will, and should, take place in stages. Each actor must feel the ability to influence the decision-making process, including generating complementary investigations in the field of safety and regarding the long-term impacts.

In the technical field of the long-term safety case for disposal it is specifically acknowledged that a safety case is built in stages, and that it should recognise, at each stage, the open issues and the relevant role of research, development and demonstration (RD&D). It is also acknowledged, in connection with the safety case, that stakeholders have a role to play as partners or reviewers and, ultimately, can help to shape the concept of safety. Specific challenges are posed to the technical community and to the roles and approaches they take.

From the FSC Topical Session on Long-Term Safety of 2007

A number of questions arose during the FSC 2007 topical session (OECD/NEA, 2008):

- How to improve on the interaction between science and society, and ensure opportunities for societal direction of RD&D?
- How to clearly and effectively communicate scientific findings and uncertainties and/or address contradictory views of process participants?
- Alternatively, how to communicate confidence in the results so far and in the process of acquiring those results?
- How to encourage "stretching" of specialists for disclosure of interests, greater transparency, broader dialogue, and to ensure rigour of research programmes?

The topical session findings on stakeholder involvement in determining the safety of an RWM facility were organised according to the phases of an iterative, analytic stepwise decision-making process. This progresses through framing, assessment and management (including its sub-phase of evaluation). Examples were given during the topical session, as follows.

Involving societal actors in framing the safety case

The topical session found that there is agreement among technical specialists on the main components of the safety case, including numerical criteria (dose, absolute risk,...) calculated for long time scales. Public risk judgments are typically qualitative. Broader issues must be incorporated into the framing: these may include concepts such as prevention, precaution, public control, justice, fairness, balance between short- and long-term protection. This incorporation of broader issues can be done through comprehensive and structured dialogue. Answers to civil society questions (e.g. on alternatives to deep geological disposal) can be sought in pluralistic hearings.

Incorporating citizen views into the assessment

The local partnership discussions on safety issues and design responses in Belgium or the community collection of data for the safety case in Nye County, Nevada are notable examples of soliciting citizen views in the course of preparing safety assessment.

The Spanish and Swedish examples from the 2005 topical session on "RD&D and Stakeholder Confidence" were also found relevant in this context. While "translation" is sometimes needed, civil society stakeholders expect that information will not be "dumbed-down". The assessment process must support access to expertise and competence-building for communities to participate meaningfully.

Handling potential conflict in evaluation

The topical session observed further that if various stakeholders evaluate radioactive waste management options, they may draw different conclusions even on the basis of identical assessment data, due to the differences in their value judgements (e.g. regarding the DBH concept in Sweden). There is a need to set prior agreement on how to handle divergences. Usually the most important requirement by stakeholders is that the process be open and transparent. In case of divergent views, it may be wise to seek the involvement of a neutral third party, who would play the role of facilitator.

Conclusions

Safety nowadays is brought about by a system of actors comprising the implementer, technical regulators, specialist groups in various advisory roles and the public, even if the legal responsibilities are different for each of the actors. The safety case should serve as the liaison amongst these three groups.

One must be aware that trust, familiarity and control are important components of safety. This is true for both the specialists and the non-specialists, even if each grouping may hold different expectations.

When the context and purpose of the study are described in the safety case, some broader issues may usefully be mentioned/discussed: *prevention, precaution, public control, justice, fairness, balance between short- and long-term protection, long-term (passive) safety*.

It is advisable to make it clearer in the safety case that, although they are not needed for arguing the intrinsic safety of the system after closure, further actions *are planned today*, such as continued oversight and RK&M preservation, that add to the system robustness both in a technical sense (reduced potential for human intrusion) and in a societal sense (continued responsibility).

It would be good if, in the safety case, the provider indicated how confidence in the results was earned and that the processes for gaining confidence were made visible.[5] Similarly, in decision making, the regulator, when issuing permits or licenses, should describe the basis for their confidence.

The NEA has elaborated a vast literature on radioactive waste governance. It would be worthwhile to analyse it for drawing further lessons from a safety case perspective.

5. This suggestion was highlighted as long ago as in the 1999 document on confidence in the safety case (OECD/NEA). Today's safety cases indeed include a "statement of confidence".

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Safety case development in the Japanese programme for geological disposal of HLW: Evolution in the generic stage

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Introduction

In the Japanese programme for nuclear power generation, the safe management of the resulting radioactive waste, particularly vitrified high-level waste (HLW) from fuel reprocessing, has been a major concern and a focus of R&D since the late 70s. According to the specifications in a report issued by an advisory committee of the Japan Atomic Energy Commission (JAEC, 1997), the Second Progress Report on R&D for the Geological Disposal of HLW (H12 report) (JNC, 2000) was published after two decades of R&D activities and showed that disposal of HLW in Japan is feasible and can be practically implemented at sites which meet certain geological stability requirements. The H12 report supported government decisions that formed the basis of the “Act on Final Disposal of Specified Radioactive Waste” (Final Disposal Act), which came into force in 2000. The Act specifies deep geological disposal of HLW at depths greater than 300 metres, together with a stepwise site selection process in three stages. Following the Final Disposal Act, the supporting “Basic Policy for Final Disposal” and the “Final Disposal Plan” were authorised in the same year.

Initial stage after the establishment of the implementing body

The Nuclear Waste Management Organization of Japan (NUMO) was established in October 2000, as a corporation authorised by the Final Disposal Act, with the remit to implement a project for the geological disposal of vitrified HLW. Based on national and international experience, which identified public acceptance as a key issue in defining the success of such projects, NUMO initiated the siting process with open solicitation of volunteer host municipalities for exploring the feasibility of constructing a final repository. This open solicitation approach was announced in December 2002 and information packages were sent to all municipalities in Japan.

NUMO will examine volunteer areas within the siting process in three stages. Before entering into the first stage, NUMO will conduct a prior confirmation of the geological conditions relating to active faults and volcanoes. If the area satisfies these geological conditions, preliminary investigation areas (PIA) for potential candidate sites will be selected in the first stage, based on area-specific literature surveys (LS), focusing mainly on the long-term stability of the geological environment. Then, in the second stage, detailed investigation areas (DIA) will be selected from the PIA, following surface-based preliminary investigations (PI) carried out to evaluate the key characteristics of the geological environment. In the final stage, detailed investigations (DI), including studies in underground facilities, will lead to the selection of a site for repository construction.

The safety case is a key factor to be considered in each selection stage. Its development, however, is a challenging process because every site requires a tailored repository concept with associated performance assessment and an individual site evaluation programme. There will thus be a number of clear decision points where sensitive choices must be made between alternative sites and associated designs. NUMO is committed to making such decisions in an open and transparent manner, which will be aided by a formal programme development process.

One of the key elements defining NUMO's safety strategy and assessment basis is the siting factors (NUMO, 2004), which determine the suitability of a site in each selection stage. Japan lies in a region of active tectonics, characterised by dynamic geological processes and events such as volcanism and earthquakes. It has to be ensured that a repository is not located where it could be adversely affected by such features. To make this clear, NUMO defined the "Siting Factors for the Selection of Preliminary Investigation Areas" (Table 1), which provide guidance and constraints for safety case development, specifically focused on geological stability in the early stage of the repository programme. Siting factors for the later siting stages will be developed and will have a similar role in future safety case development. NUMO has also developed other specific methodologies and supporting tools to manage its safety case development (Kitayama, *et al.*, 2008).

Table 1: Outline of the siting factors for the selection of PIA (NUMO, 2004)

Evaluation 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: – Geological formations – Risk of natural disasters – Hydraulic properties – Procurement of land – Geological environment – Transportation infrastructure

After the start of the open solicitation process, supported by extensive public relations (PR) activities, mayors, local council members or groups of residents in more than 10 cities/towns expressed an interest in being considered as a volunteer, the first in April 2003 and the latest in March 2009. Most of them did not enter into further consideration, due to the negative position of the governor of the prefecture where they are located, that of neighbouring cities/towns, inside/outside opposition movements immediately after announcement in the local newspaper, objections in the local council and so on.

Among them, Toyo town in Kochi prefecture applied for the literature survey in January 2007. After submission of the application, however, there was an escalation in opposition activities, including groups from areas outside the town. Responses of the town council and the prefecture were crucially negative. The mayor resigned in order to seek the opinion of the local residents, but an opponent was newly elected and withdrew the application in April of the same year.

Siting activity enhancement and demonstrating the technical feasibility of safe implementation

Reflecting the lessons learned in the case of Toyo town and others, the Subcommittee on Radioactive Waste Management, an advisory body of the Ministry of Economy, Trade and Industry (METI) which supervises NUMO's activities, proposed measures to promote the siting activities in November 2007, including:

- improvement of nationwide PR activities to promote awareness and understanding of the programme by the general public;
- improvement of regional PR activities, not only in the potential host municipality, but also in neighbouring municipalities or at prefectural level, to provide clear and accurate information on the safety of final disposal, the site selection procedure and regional development plans;
- establishment of a nomination system in addition to the volunteer siting approach, in which the national government nominates municipalities with an offer to conduct a literature survey, underpinning the greater commitment of the government to the siting process;
- proposal of potential plans for regional development based on the outreach scheme, not only in the host municipality, but also in neighbouring municipalities or at prefectural level;
- promotion of research and development and international co-operation, which helps the general public to understand the safety of geological disposal;
- promotion of collaboration between the national government, the implementer (NUMO) and the waste producers.

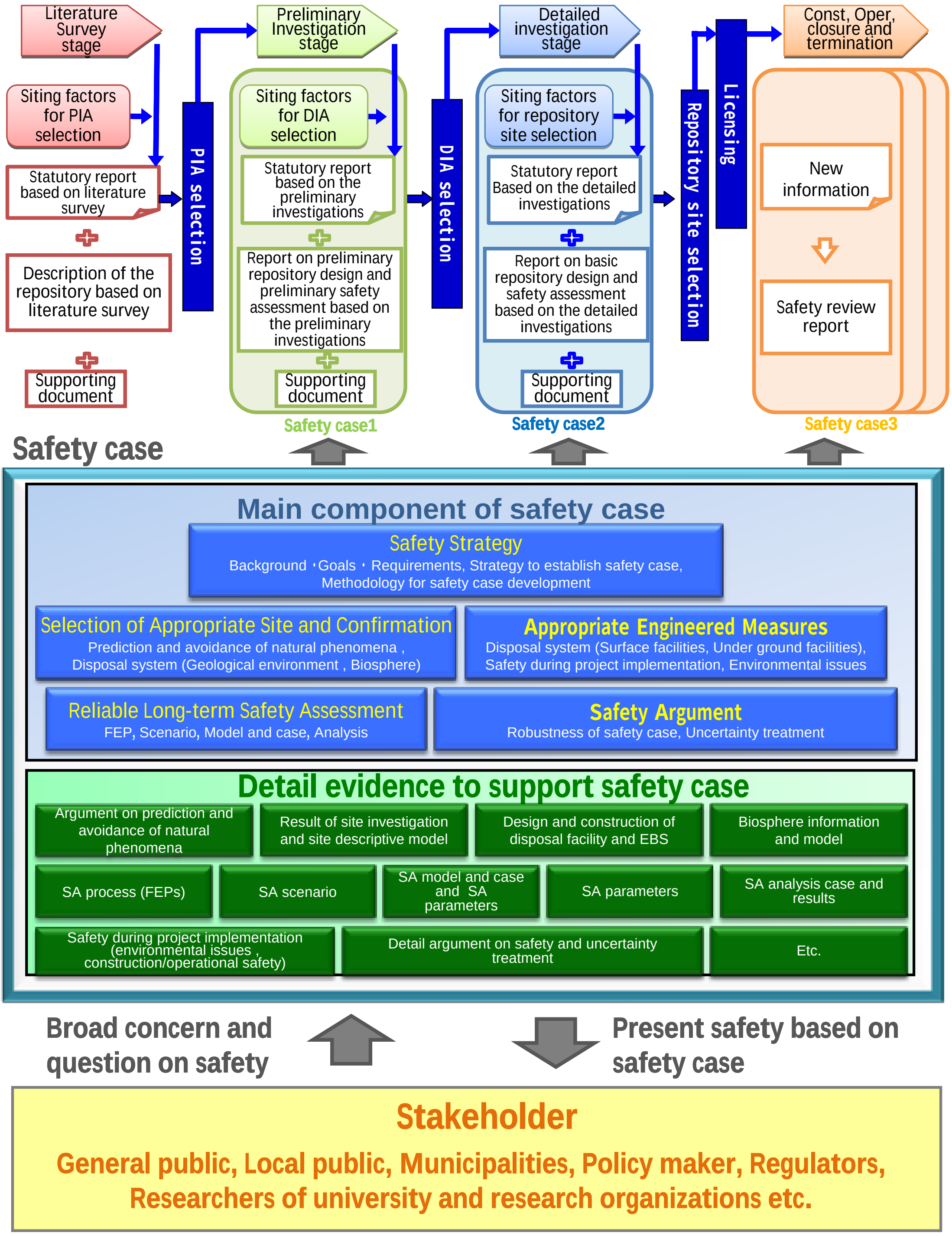
NUMO, the national government, electricity utilities and other relevant organisations took intensive actions to implement these proposals, including nationwide PR campaigns, organisation of seminars, symposia and dialogue meetings, and illustration of specific plans for regional development based on a subsidy system. The system for government nomination was set up, but has not been implemented so far.

In September 2008, the Policy Evaluation Committee of the JAEC proposed that NUMO should publish a report demonstrating the technical feasibility of safe implementation of geological disposal; the report was to be reviewed by external, independent academic institutions and revised and updated periodically to reflect state-of-the-art knowledge. In line with this proposal, NUMO issued a report in September 2011 entitled "Safety of the Geological Disposal Project 2010 – Safe Geological Disposal Based on Reliable Technologies".

In this report, NUMO describes the components and roles of its safety case as shown in Figure 1, where the safety strategy, the selection and confirmation of a suitable site, appropriate engineered measures, reliable long-term safety assessment and safety arguments are identified as the main components of the safety case. The safety case will be progressively updated and refined throughout the three-stage site selection process, licensing, construction, operation and closure, presenting and explaining safety in each stage to the stakeholders and answering their broad concerns and questions. It will integrate all the results obtained through technical activities aimed at confirming safety and providing information relevant to decision making in each stage from site selection through post-closure decommissioning of the project. These functions of the safety case are described in a confidence-building roadmap (NUMO, 2013), in which the activities for implementing one of NUMO's fundamental policies – building confidence in the safety concept – are summarised according to the project timeline.

The function of the safety case as a platform for communicating the safety of geological disposal is highlighted in a report issued by an *ad hoc* subcommittee of the Nuclear Safety Commission of Japan (NSC, a former regulatory organisation whose functions were incorporated into the Nuclear Regulation Authority in 2012) (NSC, 2011). Integration of critical issues within a transparent safety case may help to solve problems associated with communicating safety among stakeholders, such as an information asymmetry between experts and non-experts. Thus, the implementer is required to prepare a safety case addressing the concerns/questions of all stakeholders.

Figure 1: Components of NUMO's safety case and their respective roles (NUMO, 2013)



Impacts of the Great East Japan Earthquake and the NPP accident

The national government reports that the Great East Japan Earthquake (GEJE) on 11 March 2011, the associated tsunami disaster and the accident at the Fukushima-Daichi nuclear power plant have decreased the Japanese public's confidence in scientists and engineers. After the earthquake, answers to the question "Do you find scientists' statements trustworthy?", the rate of affirmative answers ("yes" and "rather yes") dropped to approximately 65%; the figure was around 10% higher before the earthquake (MEXT, 2012).

The radioactive release and contamination caused by the nuclear power plant (NPP) accident led to much concern about issues related to radiation/radioactivity and their influence on human health, including radioactive waste management. According to NUMO's annual survey, people's awareness of HLW disposal definitely increased after the earthquake (more than a 15% increase in the number of affirmative answers). At the same time, tectonic features of earthquakes and seismic phenomena such as fault movement received much attention. The role of the safety case became more important and its potential focus showed a change, emphasising the impacts of an earthquake on a repository.

In September 2012, the Science Council of Japan (SCJ) issued a report replying to the JAEC's request to SCJ in September 2010 to formulate recommendations for activities to explain and provide information on the disposal of HLW to the general public. Since the GEJE occurred during the time frame when SCJ were still assessing and discussing the JAEC's request, the related discussions were prolonged in order to assess both the impacts of the NPP accident and the evolution of national energy policy. As a result, SCJ's report presented recommendations that went beyond JAEC's request, including a fundamental revision of policies concerning HLW disposal.

The JAEC published a statement in December 2012 reflecting its considerations of the SCJ's reply. The statement expressed the JAEC's intention to maintain a policy of implementing geological disposal with renewed approaches:

- clarifying the amount and properties of HLW for disposal with respect to nuclear energy and fuel cycle policies;
- applying the latest earth science knowledge to a viability study of geological disposal and sharing the results with the public;
- improving operations according to discussions on the need and significance of interim storage;
- providing a system of sharing information on disposal technologies and the site selection process with the public;
- government's lead of the restructuring process.

Under these circumstances, the government initiated discussions on renewed approaches for HLW geological disposal in a working group on radioactive waste management (a METI advisory body) in May 2013. NUMO will provide the necessary input to the discussions.

The way forward

The situation following the GEJE in March 2011 requires careful risk communication as well as stronger trust in the implementing organisation, NUMO. Particular attention should be paid to the safety of the repository with respect to the impacts of seismic phenomena and hazards during the operational stage when the handling of the radioactive materials is carried out as in other nuclear facilities.

NUMO is planning R&D activities placing more focus on:

- the impacts of fault movement on a repository;
- the influence of earthquakes on groundwater flow profiles;
- the effects of seismic motion on the surface/underground facilities.

Hypothetical accidents during the operational stage are also being studied (Suzuki, *et al.*, 2013). The results of these R&D activities will help in formulating and enhancing NUMO's safety case in the future.

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L'expertise du Clis de Bure
(The Clis de Bure – A local commission developing its own expertise capabilities)

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Le Comité Local d'Information et de Suivi du laboratoire de Bure a été mis en place en 1999, conformément à la loi de 1991 sur la gestion des déchets radioactifs (puis la loi de 2006), afin d'assurer le suivi des recherches menées par l'ANDRA, l'information du grand public, et la concertation sur ce sujet.

Il est composé de 90 membres: des membres du Parlement, des représentants des collectivités locales (régions de Lorraine et Champagne-Ardenne, départements de la Meuse et de la Haute-Marne, communes proches du site), des représentants des syndicats de salariés, des syndicats agricoles, des organismes professionnels, des professions médicales, et des associations environnementales (parmi lesquelles les associations constituées contre le projet), des personnalités qualifiées, et des représentants de l'Etat. Sont membres à titre consultatif l'ASN et l'ANDRA.

Le CLIS s'est constitué sous forme associative et reçoit une dotation annuelle de 290 000 €, versée à parité par l'Etat et les producteurs de déchets. Il se réunit soit en Assemblée Générale ouverte au public et à la presse (3 à 4 par an), soit sous la forme restreinte du Conseil d'Administration composé de 25 membres (6 réunions par an), soit en commissions thématiques (communication, santé-environnement, localisation du stockage éventuel, réversibilité, préparation du débat public). Il organise également des débats sur des thèmes spécifiques ou des rencontres avec les habitants des communes concernées et communique via une Lettre d'information semestrielle (165 000 exemplaires), une Newsletter, des insertions dans la presse régionale ou le site internet.

Les missions dévolues au CLIS, information et suivi, nécessitent d'une part que les membres soient formés dans les domaines d'étude, d'autre part que le CLIS ait les moyens de faire appel à des experts indépendants.

Les membres du CLIS sont à la fois des relais vers la population (via les organismes qu'ils représentent ou lors des rencontres avec divers types de public) et des acteurs quand le CLIS est saisi pour avis par le gouvernement (comme pour le choix de la ZIRA ou la poursuite des recherches dans le laboratoire). Cela signifie qu'ils doivent être en mesure de comprendre et de faire partager les informations dont ils disposent.

Pour permettre l'acquisition plus ou moins poussée des connaissances dans les différents domaines de recherche (qui plus est souvent complexes), de nombreuses activités sont mises en œuvre :

- des rendez-vous réguliers avec les acteurs institutionnels du dossier (ANDRA et ASN, membres du CLIS à titre consultatif, la CNE, l'IRSN…) ;
- des visites de sites en France et à l'étranger, pour appréhender le cycle nucléaire de manière globale (Marcoule, Cadarache, La Hague, Soulaines) et avoir une vision

internationale des recherches menées sur le projet de stockage (Suède, Suisse, Belgique, Allemagne…) ;

- des formations spécifiques sur la géologie, l'hydrologie, la géomécanique, les matériaux, et tout récemment la modélisation (en deux temps : théorie avec l'IRSN, pratique avec l'ANDRA) ;
- le traitement d'aspects particuliers sous forme de débat (la sûreté, la géothermie…).

Pour bien comprendre la difficulté de la tâche, il faut se souvenir que, sauf exception, les membres du CLIS découvrent la question du nucléaire (méconnaissance originelle), et tenir compte du renouvellement régulier des membres (notamment les élus).

Le CLIS dispose également de suffisamment de moyens pour faire appel à des experts indépendants, ce qui lui permet d'avoir d'autres sources d'information et d'évaluer les programmes d'étude ou les recherches menées.

Cela peut prendre la forme d'une expertise « locale », venant de certains membres du CLIS (parmi les personnalités qualifiées, un géologue et un spécialiste de médecine nucléaire) ou d'habitants de Meuse ou de Haute-Marne ayant une formation scientifique (Mourot, Thuillier, Godinot). Dans ce cas, ce n'est pas une initiative du CLIS, mais il peut reprendre ces études pour alimenter le débat.

Cela peut passer par des analyses critiques de documents de l'ANDRA ou des recommandations sur des sujets spécifiques par des experts mandatés par le CLIS :

- avec l'IRSN dès 2000, pour évaluer le plan de suivi de l'environnement du laboratoire ;
- avec l'IEER, organisme américain, à deux reprises, en 2003 sur le programme expérimental de l'ANDRA, puis en 2011 sur les recherches ayant mené à la détermination de la ZIRA (sur ce dernier point, l'étude a entrainé des échanges avec l'ANDRA) ;
- avec les cabinets Erdyn et Toillies sur la géothermie (avec pour conséquence la modification de la campagne de forage de l'ANDRA en 2007/2008).

Encore une fois, de telles initiatives ont un coût important (elles nécessitent des appels d'offres internationaux). Elles montrent aussi la difficulté à trouver des experts qui soient jugés indépendants, et qui ne soient pas liés au projet.

Deux autres points importants pour terminer : d'une part, le CLIS doit disposer des rapports scientifiques en temps utile pour qu'une analyse critique soit pertinente et puisse éventuellement avoir des effets ; d'autre part, le CLIS doit être capable de faire connaitre le plus largement possible ces expertises.

Extended reviewing or the role of potential siting cantons in the ongoing Swiss site selection procedure ("Sectoral Plan")

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Introduction

The disposition of nuclear waste in Switzerland has a long-standing and sinuous history reflecting its complex socio-technical nature (Flüeler, 2006). Upon the twofold failure to site a repository for low- and intermediate-level radioactive waste at Wellenberg during the 1990s and 2000s, it was recognised that the respective site selections had not been fully transparent. The Swiss government, the Federal Council, accepted the lesson and, after an extensive nationwide consultation at that, established a new site selection process "from scratch": a systematic, stepwise, traceable, fair and binding procedure with a safety-first approach, yet extensively participatory. The so-called Sectoral Plan for Deep Geological Repositories guarantees the inclusion of the affected and concerned cantons and communities, as well as the relevant authorities in neighbouring countries from an early stage (Swiss Nuclear Energy Act, 2003; BFE, 2008).[1] This contribution shares experience and insights in the ongoing procedure from a cantonal point of view that is an intermediate position between national needs and regional concerns, and with technical-regulatory expertise between highly specialised experts and involved publics.

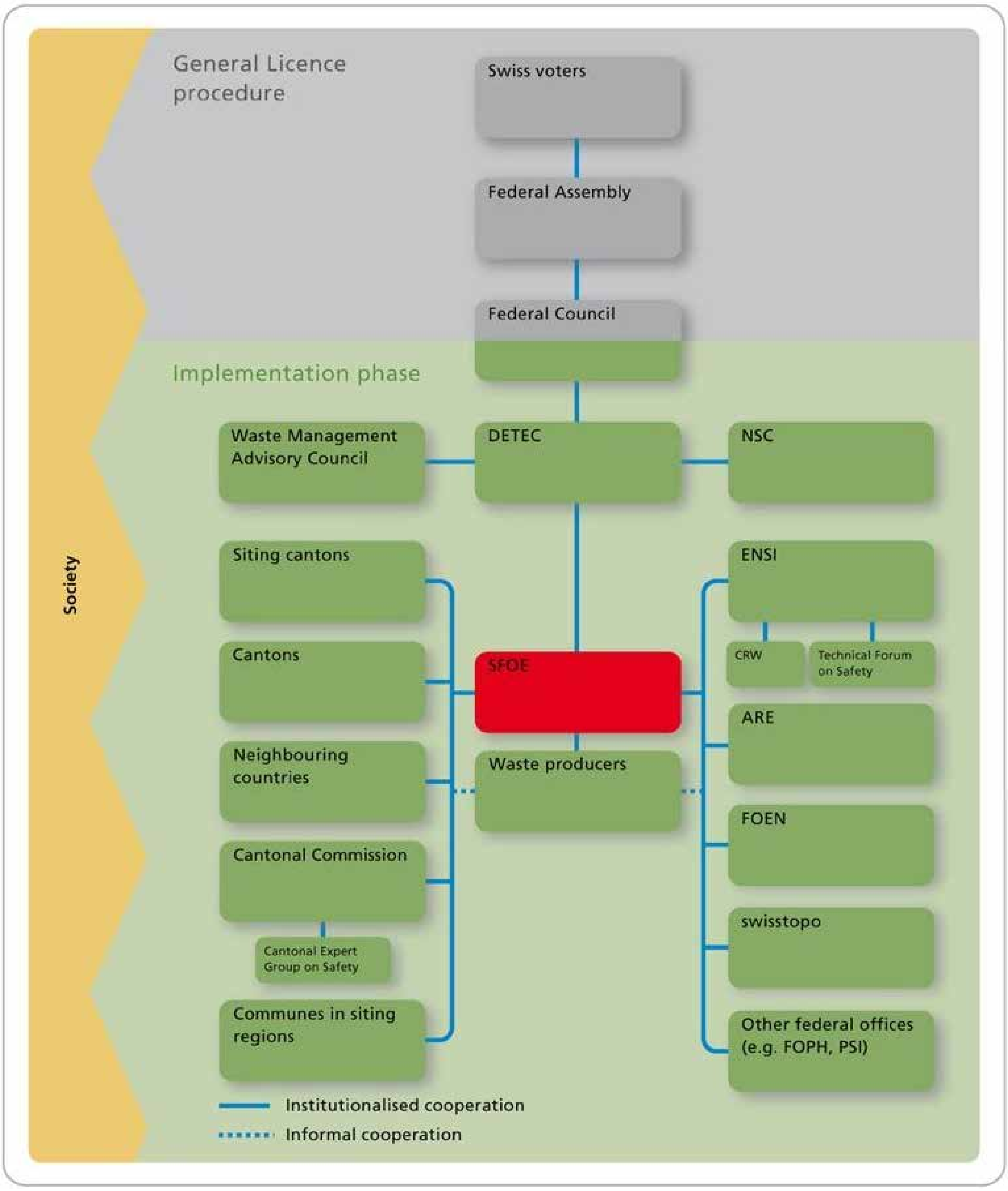
Beyond conventional reviewing: At the national-local interface

The Sectoral Plan is led by the Swiss Federal Office of Energy (SFOE/BFE, see Figure 1 and Table 1). The safety authorities and various federal commissions are responsible for reviewing and assessing all aspects relating to safety. The objective is to, within about a decade, single out a site for each repository type, i.e. spent fuel, vitrified high-level waste, long-lived intermediate-level waste, as well as low-level and short-lived intermediate-level waste. Yet, the potential siting cantons also have an important role to play.

A canton is both affected by the national issue of waste disposal and responsible for the health and safety of its residents, as well as providing them with an attractive environment (these principles are embedded in the mission statement of the Canton of Zurich, based on the cantonal constitution). Learning can take place in the interactions of all state levels (national, cantonal, regional). Even if potential siting cantons have no final say, they have some expert competence and are sought after especially in the ongoing selection procedure for technically sound yet socially tolerated repository sites, an unprecedented long-term undertaking that takes place not only in Switzerland. Even

1. See contributions by the technical regulator ENSI and the proponent Nagra in Session 3.2 for details on the three-staged procedure.

Figure 1: Principal actors as organised in the Swiss site selection procedure; explained in Table 1



Source: BFE (2008), p. 26.

Table 1: Functions, roles and activities of main actors in the site selection procedure, subnational levels highlighted in *italics*

Swiss voters	May call for an optional national referendum and thus decide on the general licence for geological repositories
Federal Assembly	Approves the general licence
Federal Council	At the end of the three stages, approves the result(s) reports and object sheets and grants the general licence
Ministry of Environment (DETEC)	Monitors and guides work on the Sectoral Plan
Federal Nuclear Safety Commission (NSC)	Advises ENSI, DETEC and the Federal Council on fundamental aspects of safety and prepares opinions on the evaluations made by ENSI in the three stages
Waste Management Advisory Council	Advises DETEC on implementing the site selection process for geological repositories
Federal Office of Energy (BFE/SFOE)	(Constitutes the) lead authority for implementing the Sectoral Plan process; prepares and updates result(s) reports and object sheets
Nuclear Safety Inspectorate (ENSI)	Reviews and evaluates the siting proposals of the waste producers from a safety viewpoint and advises the SFOE on safety issues
Technical Forum on Safety	Discusses and answers technical and scientific questions on safety and geology within the framework of the Sectoral Plan process
CRW, now Expert Group on Nuclear Waste Disposal (EGT)	Advises ENSI on geological aspects
(Federal) swisstopo	Supports ENSI on geological questions
Federal Office for Spatial Development ARE	Reviews and evaluates spatial planning aspects
Federal Office for the Environment (FOEN)	Reviews and evaluates environmental aspects
Other federal offices (e.g. FOPH, PSI)	Support the SFOE in specific technical areas [Federal Office of Public Health (FOPH), Paul Scherrer Institute (PSI)]
Waste producers	In accordance with the requirements specified in the conceptual part of the plan, search for geological siting areas and finally sites for disposal of high-level and low-/intermediate-level waste, evaluate these sites and propose that they be integrated into the plan; responsible for preparing and submitting the general licence application together with the necessary supporting documentation
Potential siting cantons	Work together with the federal government and support it in carrying out the site selection process; co-ordinate the procedure for modifying the cantonal structure plans and ensure co-operation with the (communities) in the siting region
Cantons	As part of the official hearing process, express opinions on drafts of the results reports and object sheets and participate in the process as specified in the Nuclear Energy Act and Spatial Planning Act
Cantonal commission (Committee of the Cantons)	Ensures co-operation between government representatives of the siting cantons and affected neighbouring cantons and countries and supports the federal government in implementing the selection procedure
Cantonal Expert Group on Safety	Supports and advises the cantons in evaluating safety-related documentation
(Communities) in the siting regions	Work together with the SFOE in organising and implementing regional participation and represent regional interests
Neighbouring countries	Express opinions on the results reports and object sheets as part of the hearing process and participate (according to Swiss regulations and joint agreements)

Source: BFE (2008), p. 27 *passim,* adjusted.

though representatives from the potential host regions sit in technical committees of the Sectoral Plan, they have to trust the upper levels. In contrast, the cantons are resourced with experts and have close access to the (making of the) safety case. It is also their duty that participation on the regional level is more than just a buzzword or sophisticated public relations (Stauffacher, 2012). Thus they may be intermediaries between national deciders/experts and local affected/concerned laypersons (Flüeler, 2012) and help transcend the traditional roles of experts vs. the public by adopting a national commitment as well as a regional embedding (OECD/NEA, 2012b). By way of this concept, reviewing goes beyond traditional (peer) reviewing and can considerably augment the "statement of confidence" required by the safety-case methodology; it even has the potential to enhance credibility in the safety case (OECD/NEA, 2013, p. 9], "[convince]...all relevant groups...the public...of the adequacy of the analysis" (OECD/NEA, 2012a, p. 81), even foster trust in the – wider – waste community.

The cantons work together with the federal government and provide support in implementing the site selection process, collaborating with the regions and communities and modifying their cantonal structure plans. The Sectoral Plan requires them to act in the place of regions in case a region refused collaboration (which has not occurred so far). A cantonal commission (Committee of the Cantons) was established to ensure co-operation among the government representatives of the siting cantons and the concerned neighbouring cantons and countries. The commission also makes recommendations to the federal government. It set up an independent Cantonal Expert Group on Safety to get advice when evaluating safety-related issues. The Technical Forum on Safety with all major players, including representatives of the potential siting cantons and regions, discusses and answers questions on safety and geology received from anywhere, such as the public, the communities, siting regions, organisations, cantons and public entities in neighbouring countries.

Beyond conventional proof of safety: The "hybrid" level to integrate different views and interests

Even though the nuclear community has long recognised that the required long-term safety of repositories "is not intended to imply a rigorous proof of safety, in a mathematical sense, but rather a convincing set of arguments that support a case for safety" (OECD/NEA, *et al.*, 1991, p. 11; OECD/NEA, 1999, p. 10 *passim*), it has been difficult to truly internalise and "live" its socio-technical nature (OECD/NEA, *et al.*, 1991; OECD/NEA, 2004). The famous question of "How safe is safe enough?" (Fischhoff, 1978) cannot be answered technically because it is a political question. Thus, albeit the waste problem is inherently driven by technology and, indeed, a technological constraint, in the end, it has to be solved by society (Flüeler, 2001). In such a situation, a "hybrid" level – on which the cantons are in the nuclear waste case – lends itself to integrate or, at least, bring the technical and the social-political worlds together. The fact that disposal is seen as a national duty assigns them quite a portion of responsibility (some of them even are shareholders of electric supply companies with nuclear power plants producing the bulk of the waste). As they are in charge of spatial planning according to the Federal Constitution (1999, status as of 2012), it is also their duty to integrate different concerns and subjects, and that supra-regionally and in the long run.

Authentic guardian of a transparent and open outcome procedure

Nuclear issues have internationally been contentious and still suffer from a traditional decide–announce–defend paradigm of decision making (Kemp, 1992), traceable to a historically rather secretive environment and technocratic approach of the "nuclear establishment" (Jacob, 1990, p. 21 *passim*). Unlike other controversial technical issues, "nuclear waste policy was", in Jacob's words, "not the engine that drove politics, but the product of political, economic, and social engines which drove the politics of nuclear waste." Furthermore, non-experts (have to) use availability heuristics to assess risk issues; many anchor the issue of nuclear waste, new and unfamiliar to them, to military and/or civil management of (legacy) waste, weapon production, or even bomb testing (Flüeler, 2006, p. 67 *passim*, p. 119 *passim*).

Against this background it is vital to respect and address the underlying critical attitude towards nuclear waste, especially when dealing with the very complex demonstration of long-term safety. For this reason the cantons have always emphasised the "process product" characteristics of the Sectoral Plan: the sense of responsibility to help find technically suitable and socially tolerable locations, based on a safety-first yet open-outcome approach, transparency, traceability and a questioning attitude. Such an attitude has lent itself as helpful and responsive in view of the recent point of criticism raised that the federal bodies suffered from "regulatory capture" supposedly by the proponent (Buser, 2012).[2]

Substantively, the cantons have been able to make various contributions to the safety case so far, for example (all publicly available at www.radioaktiveabfaelle.zh.ch > Ausschuss der Kantone):

- Stage 1 (to find geologically suitable siting areas in Switzerland): Factual comparability of the siting areas has been demanded as a prerogative for selection as the ones proposed by Nagra lie in seven different cantons and only one had been explored by 3-D seismics accompanied by a reference borehole in the project demonstration of disposal feasibility in 2002 (Nagra, 2002) The cantonal expert group recommends that host rocks be solely rejected on grounds of a robust – and comparable – knowledge base, or, inversely, all potential siting areas be kept until the remaining uncertainties are clarified through dedicated investigations (AG SiKa/KES, 2010, p. 10]).
- Stage 2, underground programme (to select at least two sites for each repository type, ongoing): In order to achieve the recommendations from Stage 1, Nagra agreed to carry out 2-D seismics for all siting areas with heterogeneous host rocks (the interpretation is expected for winter 2013/14), and ENSI accepted to introduce so-called "stopover" sessions (AG SiKa/KES, 2011) to closely follow up on the progression of Nagra's knowledge base. The methodology hereto was jointly developed with the cantonal experts (ENSI, 2012a, 2012b), to specify and put on a set timeline the so-called provisional safety analysis and the safety-related comparison (ENSI, 2010) according to the requirements (HSK, 2007) previously proposed by the technical regulator (then called HSK, now ENSI) and stated in the Sectoral Plan (BFE, 2008, p. 40, 51 *passim*). Only if the geological models and the other data base are sufficiently reliable and robust, Nagra is allowed to, at the end of Stage 2, submit the necessary documents to finalise this stage.
- Stage 2, above-ground programme (ongoing): In parallel with the geoscientific investigations, a selection process is underway to find a suitable site for the surface installations of the respective repository (such as the fuel packaging facility in case of the high-level waste repository), and that in each potential siting region. Originally, it was meant to be a regional-planning exercise only, together with the concerned and affected communities and other stakeholders under the sail of "regional participation" (BFE, 2008, p. 82). When the issue came nearer it became clear that the surface facilities, the access structures and the repository

2. According to separate investigations both the Federal Ministry of Environment and the ENSI Board came to the conclusion that neither SFOE nor ENSI have been taken in by any player in the Sectoral Plan (www.uvek.admin.ch/dokumentation > Verfahren des Sachplans, 3 December 2012; www.ensi-rat.ch > Externe Abklärung, 3 December 2012).

underground have to be assessed in an integrated manner (for example, the appraisal of thick groundwater layers as long-term reservoirs for drinking water). Technical rounds were carried out, the cantonal experts issued a respective report (AG SiKa/KES, 2012), and there is an intense public debate on ramps vs. shafts. This concern was recognised, and Nagra has to carry out a risk assessment of the access structures with regard to construction and (also long-term) operation issues (ENSI, 2012c).

All in all, many involved cantonal civil servants and experts are in intense scientific and technical discourse on the national level; they also advise and (in higher-level planning issues) somewhat lead the regional stakeholders, this in policy rounds as well as technical sessions of the respective regional committees (on surface facilities, on safety and on socio-economic issues). The site selection process is and continues to be a long and tough slog, but it is ongoing, slowly but surely, there are no major dropouts, critical comments were recognised, the work has been improved. And the cantons are willing to be guardians of a transparent, fair process with an open outcome as well as a safety-related and socially tolerable final product, the location for safe and accepted final repositories for nuclear waste (where construction, operation and regional embedding will have to take place in the steps to come).

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Poster Session

The marker: A precious link between generations and a part of the long-term safety story

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Introduction

High-level radioactive waste brings us face to face with a social, emotional, ethical, political and environmental situation at the heart of which lies the security of the living world.

From now on, mankind has to make commitments to protect itself. Building an artistic device, a continuous creation that links and reveals the situation will inform and shed some light on the objectives.

In co-operation with industries that manage geological repositories, the installation of markers above ground introduces non-technical aspects that can increase safety.

In the art world, many works of art produced throughout the 20th century associate art with waste as vestiges for keeping!

The use of waste has shaken the nature of art since Marcel Duchamp, the eye of Man Ray as well as the definition of the artist. The waste recycling industry has modified the way we see it.

For what is waste? It embodies above all the imperfect and also a “time capsule”.

In the case that concerns us, waste stays buried in our doubt, secured by our current financial means that are substantial. The notion of time for its decay is calculated and, at the end, it vanishes.

Its visibility is next to nil, whereas its presence remains very powerful and thrills the imagination. The hosting community is permeated with it.

At the same time the laboratory at the disposal facility provides economic dynamism, an analysis of upheavals in this territory, an exhibition of artistic devices and an awareness of danger.

Passing on memory

Humans are passers of memory and, for several generations, on the surface of the site, they express their doubts, their emotions, their fears: the expression of the survival instinct on earth.

Markers are a transmission medium for memory of sites through the implementation of a specific architectural vocabulary: code, symbols and signs which evolve with time.

This artistic device will remain contemporary to the sites and those who follow us, thanks to a unique *modus operandi.*

The markers

Repositories as platforms for artistic research

The markers are multi-layered artistic devices built at the land surface that create a visible framework related to cultural and social values for these archives of the future.

The marker, tangible at the outset and becoming an imaginary or immaterial picture in remote time, leaves an area not to visit, the danger to life will have another meaning.

This implies a higher level of mindfulness, as “artists-guardians”, “writers-explorers” or “scientists-archaeologists” involved in this creative laboratory provide multidisciplinary dynamism.

A new kind of sensitive monitoring will be made possible by intensifying a community-based and interdisciplinary dialogue between fields that run parallel to each other.

The development of the marker must be drawn slowly until the project sets its own rules that must be complied with. The research process should be carefully conducted because tied to the idea of unique laboratory for memory and safety.

Art acts as a transmitter of an archive, through the implementation of an architectural vocabulary.

Art enhances the level of vigilance concerning the danger through the symbolism of this marking.

This *modus operandi* is a continuous artistic work on site in an experimental laboratory, open to various disciplines that enrich knowledge.

The regeneration of markers over time is possible thanks to new techniques, materials and the evolution of the perception of the danger of radioactivity to the living world.

“Art emerges out of particular situations and I believe that artists can only benefit from what their environment sends to them.” – Al Anatsui, African artist

Proposal for markers projects

Having a long-term vision:

- About the *territory*:
  - parameters and constraints.
- A *guidance* nurtured by content that is:
  - theoretical;
  - scientific;
  - artistic;
  - philosophical;
  - environmental and human.
- *Feature*:
  - understanding the objectives by site survey.
- In *co-operation*. Reading of:
  - the sketches;
  - the pre-project phases;
  - the markers projects.

- *Criteria.* The ability of the marker to:
  - awaken in the long term in a memorial and security perspective;
  - be regenerated over time.
- *Planning:*
  - implementation;
  - inclusion of new contributions;
  - budgeting;
  - estimation of the impact on the area.
- *Means:*
  - European and international exchanges through meetings to develop a nuclear culture;
  - the appointment of curators.

The various aspects of safety

Safety is a major goal that can be achieved through current technologies, various tools, such as markers, networks, satellites, etc.

Tomorrow, sensors embedded in markers at the periphery of the sites will increase safety and make the current situation legible to all.

Maybe in a thousand years, in another time, nothing.

Further to the striking facts observed in the world, and later, with a more acute awareness of the strengths and wave effects that govern us, the nuclear culture will become part of our lives.

The landscape is a casual collection of fragments that "give" to each other, slowly shaped by the nature itself of the world, by man, plants, forces, the wind, the sea,... To apprehend the geological storage and report on it, one must move through, measure the presence, stay in sight, take some time to read the history of this landscape.

One must also be aware of the severity of toxicity, the recklessness of the nuclear operators, for not having a long-term questioning on the topic, to avoid being blind again.

With markers projects, we aim to convey and enhance safety.

Understanding markers, step by step, is:

- looking at how the light rays hit them and reveal them;
- raising the question on the symbolism of the colours chosen, materials' shapes, their position in space and what they transmit to us;
- wondering why the ground surface was made uneven.

The site is slowly becoming, in parallel to storage, a potential platform for "landscape design" where monuments of a new kind appear, that are dedicated without pathos for modification. Their presence indicates a memorial desire to remember, the attention to danger and safety and entering a form of culture, the nuclear culture: a new way of commitment.

Issues

The specific nature of markers is not the only issue.

It participates in the creation of a new type of archive of our society. It demonstrates our ability to awaken in European civil rights. It engages with knowledge and information management, to enhance education about this subject.

It introduces the artistic and philosophical factors.

By the study of an unprecedented way of communicating, the marker passes on memory, questions about the danger in a territory and constitutes a safety barrier.

This obliges us to reconsider the notions of perspective, heritage and knowledge, materials, art, time, technology, co-operation and security.

Communication

The repeated presence of objectives to local associations, schools, universities, regional museums, audio-visual media, the press, on social networks, scientific or artistic magazines, in exhibitions, symposiums, call for projects, European and international institutions, ignites and broadcasts a project. This evokes the question of knowledge and therefore safety.

The activity of the laboratory is not factual but continued, it asserts its societal role.

When talking about markers, or archisculptures, the words “art” and “artworks” come first to mind.

In the Western world, we are still prisoners of many ideas from the 18th century, where the first museums have an idealistic vision of art and are seen as “temples”.

Here, no museum, but scientists who advise management agencies to handle the archaeological stratum of the 21st century in the nuclear countries. It examines and exhibits geological storage and safety.

On the other hand, this new stratum calls the philosopher, the sociologist, the artist because the lifetime of the atom takes human beings back to their own time scale and the geological disposal facility questions it thoroughly.

Today, in Europe, this type of place appears far from urban landscapes. With a larger density of population, cultural differences that distinguish rural and urban activities and regional peculiarities could slowly fade away.

The issue remains: The security of the living world.

Generally, in architecture or landscape design, one looks for the exception to create the event. Political power and economic models have their own requirements, sometimes forgetting about the primary objectives.

Throughout history we can observe the striking links between power and artistic systems and, still multiplying today, just like pictures, they are exported as a label.

On the topic at hand, the aesthetic qualities should not remain riveted to our Western standards but be specific to the transmission of the memory of a peculiar aspect of our industrial activity.

The marker must be read by all for the safety of all; “all” are the project recipients.

We are not in an outdoor museum, we are for a while on a lively platform looking on the marking of a place containing the most toxic waste man has ever generated.

There is no purchase of works, work is continuous, unsigned, it is the memory and the security of the living world that is at the heart of the ambitions.

Conclusion

The proposals that are described here are the result of a decade of research.

It seems that from the beginning there has been negligence on the part of the nuclear industry regarding the storage of high-level radioactive waste.

The general public is not very knowledgeable about the subject. Projects for geological disposal often give the local population the feeling of an ablation of a piece of territory.

To sum it up, in the long run, we consider the study of two barriers for the safety of high-level waste repositories, two approaches that will continue to develop in parallel:

- technological and industrial resources;
- the implementation of markers on the surface.

To address the economic, social, environmental and political challenges, one must raise the reflections that are relevant to areas that are linked to humans themselves, the link, makers of memory, securely tightened in their own contradictions.

Uncertainty in hydraulic tests in fractured rock

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Abstract

Interpretation of hydraulic tests in fractured rock has uncertainty because of the different hydraulic properties of a fractured rock to a porous medium. In this study, we reviewed several interesting phenomena which show uncertainty in a hydraulic test at a fractured rock and discussed their origins and the how they should be considered during site characterisation.

Introduction

Because groundwater in fractured rock tends to flow through fractures rather than rock matrix due to their different hydrogeological properties, groundwater flow in fractured rock can be characterised by heterogeneity and discontinuity, introducing uncertainty in characterisation, conceptualisation and simulation of a groundwater flow in fractured rocks. Especially, the analytical solutions for interpreting hydraulic tests are generally induced from the assumption of a porous medium, and the estimated hydraulic properties of a fractured rock are likely to have uncertainty.

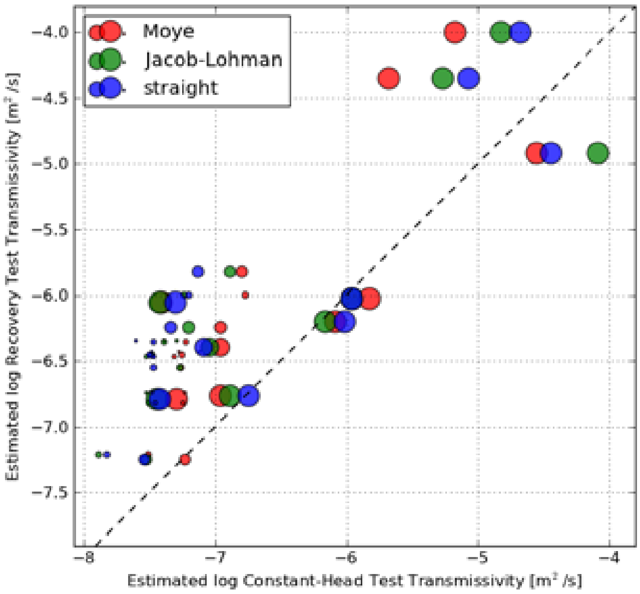
Since the Korea Atomic Energy Research Institute Underground Research Tunnel (KURT), whose host rock is granite, was constructed, numerous hydraulic tests have been conducted at the installed boreholes in KURT. From the hydraulic test results, several interesting phenomena were observed and two of them were reviewed in this study in the view of uncertainty in hydraulic tests.

Cases

At the DB-1 borehole in KURT, which is a vertical borehole 500 m in length, constant head withdrawal and recovery tests were conducted at several packed-off sections. Please note that the constant head withdrawal test is the pressure decreasing test and the following recovery test is the pressure increasing test. Figure 1 shows the comparison between the estimated transmissivities from the constant head withdrawal and recovery tests. Although the estimated transmissivities were similar to each other in the fracture zones, the constant head withdrawal test results were approximately an order of magnitude smaller than the recovery test results. A series of field-scale *in situ* tests indicates that this phenomenon originated from the effect of fracture aperture change due to water pressure change and the hydraulic head change during a hydraulic test leads to a change in a fracture aperture (Ji, *et al.*, 2013). In the tests, the aperture of a

Figure 1: Comparison between the estimated transmissivities from the constant head withdrawal and recovery tests

The size of the circles is the fracture density of the test interval



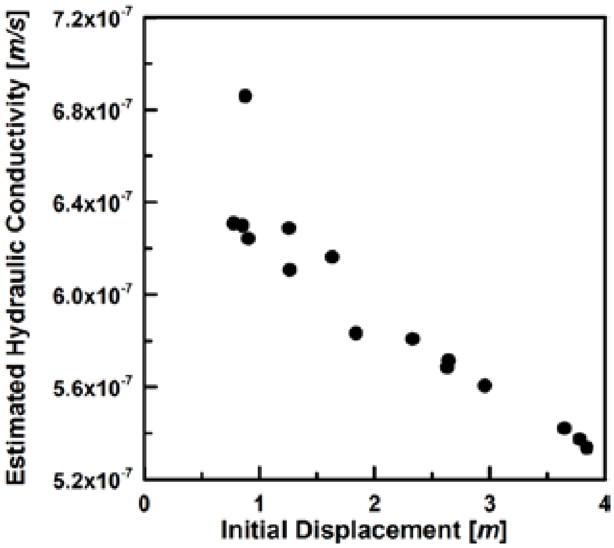
fracture in the test zone increased by factors of 1.22~1.44 when water pressures of 2~5 bars were applied with the initial water pressure of 0.1 bars. The results show that a small change of an aperture induces considerable changes in the estimated fracture hydraulic parameters because the fracture transmissivity is proportional to the cube of the aperture and the tortuosity in a fracture is also influenced by the aperture.

A slug test is one of the hydraulic tests where the hydraulic properties of an aquifer are estimated using the monitored recovery after a sudden displacement of the hydraulic head in a borehole. At the TB-5 borehole in KURT, which is a vertical borehole with a length of 30 m, a series of slug tests were conducted at several packed-off sections. Figure 2 shows the estimated hydraulic conductivities under various initial head displacements and the estimated hydraulic conductivity decreased as the initial displacement increased. The analysis of the test results indicates that the non-linear groundwater flow had been created during the slug tests with large initial head displacements leading to an underestimation of the hydraulic conductivity (Ji and Koh, 2013).

Concluding remarks

Our results show that the estimated hydraulic parameters of a fractured rock from a hydraulic test are associated with uncertainty due to the changed aperture and non-linear groundwater flow during the test. Although the magnitude of these two uncertainties is site-dependent, the results suggest that it is recommended to conduct a hydraulic test with a little disturbance from the natural groundwater flow to consider their uncertainty. Other effects reported from laboratory and numerical experiments such as the trapping zone effect (Boutt, 2006) and the slip condition effect (Lee, 2014) can also introduce uncertainty to a hydraulic test, which should be evaluated in a field test. It is necessary to consider the way how to evaluate the uncertainty in the hydraulic property during the site characterisation and how to apply it to the safety assessment of a subsurface repository.

Figure 2: Estimated hydraulic conductivities under various initial displacements



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Microbial processes need to be addressed in a competent safety case

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Microbial processes

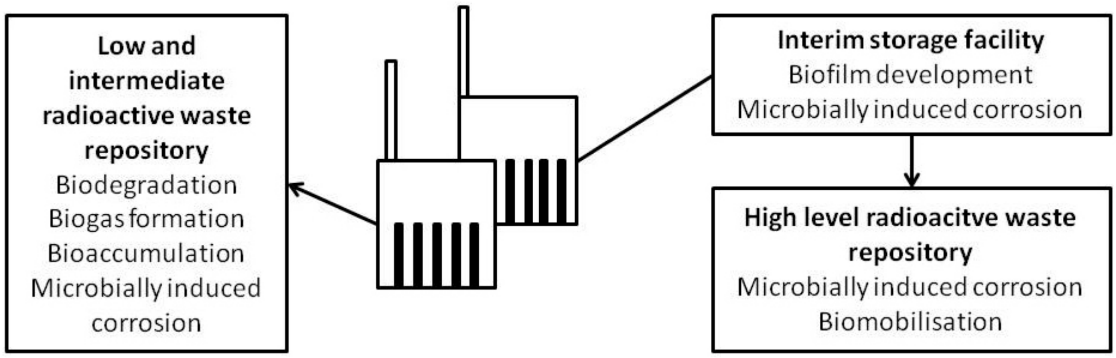
Biological life processes should be separated from chemical inorganic processes. This is because most life processes are running along biochemical process pathways that are very different from those of inorganic chemical processes. Many chemical processes are restricted by various reaction barriers, e.g. the reduction process of sulphate to sulphide (Cross, 2004; Goldstein, 1994). They are, therefore, very slow or not possible at all under conditions typical for repositories for interim storage and long-term storage of radioactive wastes. Biological processes, consequently, include many reactions that do not occur in sterile, lifeless chemical systems. This is because life has the ability to over-run activation energy barriers and other energetic circumstances that block spontaneous chemical reactions. Life is possible from -20°C up to around 113°C, where all life processes stop. Life is also possible within a large pH range, from pH 1 up to at least pH 12.5 (Pederson, 2004; Takai, 2011). Finally, many processes overlap biology and chemistry such as sulphide and ferrous iron oxidation, which occur both as chemical and biological processes. Life processes in radioactive waste interim and long-term repositories will almost exclusively be under the control of micro-organisms.

Microbial processes comprise red-ox reactions with decomposition and the production of organic molecules with different electron donors, energy sources and electron acceptors. Organic carbon in wastes, buffer materials and reduced inorganic molecules such as H_2 from anaerobic corrosion processes and methane from geological sources are possible electron donors and energy sources for microbial processes. During the microbial oxidation of these energy sources, micro-organisms preferentially reduce electron acceptors in a particular order. First O_2, and thereafter nitrate, manganese(IV), ferric iron, sulphate, sulphur and carbon dioxide are reduced. Simultaneously, fermentative processes lower pH and supply the metabolising micro-organisms with, for example, H_2, methane and short-chain organic acids such as acetate. Fermentation, in contrast to respiration, does not require an external electron acceptor, the oxidation-reduction process comprises rearrangement of electrons in exogenic modes, thereby releasing energy for life processes.

It is well known that microbes can mobilise trace elements (Anderson, 2011, and references therein). Firstly, unattached microbes may act as large colloids, transporting radionuclides on their cell surfaces with the groundwater flow. Secondly, microbes are known to produce ligands that can mobilise soluble trace elements and that can inhibit trace element sorption to solid phases. Thirdly, viruses are commonly present in large numbers in groundwater and most of them are the size of colloids and their proteinaceous shells readily sorb trace elements (Kyle, *et al.*, 2008a, 2008b). Fourthly, a large group of microbes catalyse the formation of iron oxides from dissolved ferrous iron in groundwater that reaches an oxidising environment (Anderson and Pedersen, 2003). Biofilms in aquifers will influence radionuclide retention processes in groundwater (Anderson, *et al.*, 2007).

Radioactive waste repositories will not be sterile, and microbial processes will occur at rates determined by the prevailing chemical and physical conditions of the respective repository (Figure 1). Low- and intermediate-waste repositories (LIWR) will contain large amounts of organic material that can be degraded by micro-organisms with formation of gas and compounds that may mobilise radionuclides. Interim storage in pools with pure water must be applied to cool down high-level radioactive wastes (HLRW) before final disposal. Some micro-organisms have adapted to live at very low concentrations of organic carbon (<2-3 mg organic carbon per litre). These microbes often live attached in biofilms and they increase the risk for clogging, accumulation of radionuclides on surfaces and biocorrosion of stainless steel. In deep repositories for HLRW, micro-organisms in groundwater and buffer materials can produce sulphide and other substances that are corrosive to metal canisters. Some microbes may mobilise radionuclides. Below follows a brief compilation of various aspects regarding microbial processes than must be addressed in safety cases.

Figure 1: Major microbiological processes that can occur in radioactive waste storages and repositories



Interim storage facilities

Growth of micro-organisms are often observed in interim storage facilities (ISF) (Masurat, 2005; Wolfram, 1997). Microbial growth in ISF is a concern for ion exchanger and rod filter clogging. Biofilm can also bind radioactive substances and particles, thereby accumulating radioactivity at positions in the systems supposed to be radiation free. Biofilms are known to increase the risk for microbially induced corrosion (MIC) which eventually may damage stainless steel construction in storage pools and tubes for circulation of cooling water.

Low and intermediate radioactive waste repositories

The pH in low and intermediate radioactive waste repositories (LIRW) will decrease over time (Small, 2008). Therefore, the influence of microbial processes will increase in magnitude concomitantly. In an aerobic repository environment microbial degradation of plastic polymers via de-polymerisation will occur. The initial degradation is mediated by de-polymerases which break down the long polymer into oligo-, di- and monomers (Gu, 2003, and references therein). The closer to a natural polymer the structure of the polymer is, the faster the microbial degradation proceeds. The aerobic degradation end products are carbon dioxide and water. Microbial degradation of polymers occurs in anaerobic environments as well. The degradation products in anaerobic processes are organic acids, carbon dioxide, methane and water (Gu, 2003, and references therein). The degradation of polymers is often a slow process but initially when O_2 is present microbial degradation will produce smaller units that can be utilised in anaerobic microbial degradation processes. Polyethene is degraded by lignin degrading white rot fungi by the action of a manganese peroxidase (Iiyoshi, 1998). Especially in nutrient-limited conditions the elongation capacity and tensile strength of the polyethylene were found drastically decreased by degradation of the fungi tested.

Bitumen is a complex colloidal system consisting of a mixture of mainly high molecular aliphatic and aromatic hydrocarbons. It consists mainly of four different compound groups; saturated hydrocarbons, cyclic hydrocarbons, resins and dispersed particles known as asphaltenes. Roffey and Nordqvist (1991) concluded that biofilm formation on bitumen occurred both under aerobic and anaerobic conditions. A pH of 9.8 did not inhibit growth on bitumen by aerobic micro-organisms. Degradation studies of bitumen have shown that parts of the hydrocarbons in bitumen are biodegradable. Potter and Duval (2001) could measure a 50% decrease in the aromatic and aliphatic fractions in a bitumen-based fuel sample (trade name Orimulsion). The bitumen had a large surface area, glass beads were covered with the substance and the degradation was aerobic. Factors that affect the degradation rate are surface area, temperature, availability of additional nutrients like nitrogen and phosphorous and of course the access to O_2. Bitumen mixed with waste will have an uptake of water. This happens both when bitumen is immersed in water and placed in humid air. Tests showed that samples with a mixture of bitumen, salt and sludge particles swelled about 10-15% in one year and samples with radioactive waste swelled to twice its original volume. Other tests showed no difference between non-radioactive and radioactive waste samples (Pettersson, 2011). Water in the bitumen-waste mixture will increase the possibility for microbial degradation of the hydrocarbons in the bitumen concomitant with degradation of the waste product, because water is needed for microbial processes to proceed.

Cellulose is a compound that is easily degraded by microbial processes. It is also chemically degraded under alkaline conditions (Glaus, 1999; Van Loon, 1997, 1999) to compounds that can be further degraded by microbial processes (Baily, 1986). There will be some moisture in the material which will facilitate heterotrophic degradation by mould and bacteria especially as long as there is O_2 present in the repository. During aerobic respiration water is formed which enhances further degradation of the cellulose material. When O_2 has been consumed, fermenting processes are likely to start (Baily, 1986). These processes acidify the environment through production of acids like acetic, citric, oxalic, together with carbon dioxide. In addition, anaerobic respiration processes can occur such as nitrate-reduction, iron-reduction and sulphate-reduction depending on how the ion-exchange resin was prepared, for instance with Na salts like nitrate or sulphate. Lignin is one part of the carbohydrates in wood. This compound is mostly degraded by fungi named “white rot”.

The ion-exchangers used in nuclear power plants are usually strongly acidic with styrene resin. One type of ion-exchange is Amberlite IR-120 which is a strong acidic ion-exchanger with sulphonic acid groups on a styrene resin (Pettersson, 2001). Styrene (vinyl-benzene) is a polymer and it is a natural component in plants. It is aerobically degradable by different types of bacteria (Grbic-Galic, 1990; Omori, 1974, 1975; Shirai, 1979; Sielicki, 1978). It has also been shown to be degraded by an anaerobic consortium of micro-organisms (Grbic-Galic, 1990). Ion exchange resins can swell up to 200% in water (Snellman, 1985). The ionic state of the resin affects the swelling capacity, H^+ and OH^- results in the largest swelling. Resins are therefore treated with sodium sulphate to reduce the swelling. Glass-wool and mineral-wool are examples of insulation materials that are deposited in LILW repositories. During the production of the insulation phenol-plastic (Bakelite) is used as binding agent. From this addition phenol and formaldehyde are produced in the insulation products. These compounds are organics that can be degraded by micro-organisms both aerobically and anaerobically (Flyvbjerg, 1993).

Microbial growth is possible in a waste matrix that has been solidified with cement or concrete. The main microbial process that can degrade concrete is acid production by micro-organisms. Concrete degradation by micro-organisms is a well-documented problem in sewage pipes made of concrete and it has also been found in LIRW (Gorbunova, 2012). Because of the high load of organic material in many LILW, the O_2 in the water is rapidly depleted by aerobic micro-organisms and anaerobic degradation by sulphate-reducing bacteria (SRB) takes over with production of hydrogen sulphide. The hydrogen sulphide is oxidised if it contacts O_2 and sulphuric acid is formed. The hydrogen sulphide oxidation can also be anaerobic by sulphur-oxidising bacteria using nitrate as electron acceptor (Preisler, 2007). This occurs at lake/sea sediment surfaces and i.e. by the sulphur-oxidising bacteria *Beggiatoa*, which is responsible for the oxidation of hydrogen sulphide in marine sediments.

Studies have been performed on microbially-induced degradation of concrete. Small, *et al.* (2008) investigated the biogeochemistry and gas production from low- and intermediate-level waste in concrete and steel tanks. During the seven years of the experiment, the measured pH in the storage decreased from between 10 and 11 to 7.5 in the water above the concrete tanks used for storage of different kind of waste. In tanks with biodegradable material, the pH was as low as 5.5. Vincke, *et al.* (2002) studied the influence of different additives on the hydrogen sulphide oxidation and sulphuric acid production. In their study, in laboratory scale experiments, the pH decreased from 8 to 1 in microbially active systems. In sterile systems the pH did not change. Micro-organisms capable of causing degradation of silicate materials were found in a LIRW that had been in operation for 15-45 years (Gorbunova, 2012).

High-level waste repositories

A large and diverse array of investigations and experiments has been conducted to increase the understanding of microbial processes in deep groundwater and HLRW repositories. Sampling procedures have been developed and thoroughly tested, as have underground facilities for model studies (Nielsen, 2006; Pedersen, 2012a, 2013). The first important parameter to analyse in repository environments is biomass. Three different methods have been developed and found to correlate. Microscopic counts and biochemical analysis of adenosine-three-phosphate (ATP) agreed well (Eydal, 2007). The determination of cultivable micro-organisms and ATP also agreed well when analysed (Pedersen, *et al.*, 2008). Many different phenotypes of micro-organisms have been found via cultivation (Hallbeck, 2008, 2012), including fungi (Ekendahl, 2003; Reitner, 2005). The influence of viruses on microbial processes has been identified as an important factor to include in model studies. In particular, they seem to have an important mitigation effect on sulphide production by SRB (Eydal, 2009; Kyle, 2008a; Pedersen, 2013)

Microbial biofilms was found to significantly influence the sorption of radionuclides on glass and rock surfaces (Anderson, *et al.*, 2006a, 2007). In addition, it has been demonstrated that microbial iron oxidising biofilms are strong sorbents for trace elements (Anderson, *et al.*, 2006b; Anderson and Pedersen, 2003). Micro-organisms from deep groundwater produce complexing agents that mobilise radionuclides (Essén, 2007; Johnsson, 2006). Such complexing agents have a strong influence on radionuclide mobility. They can mobilise uranium (Kalinowski, 2004, 2006) and strongly bind curium (Moll, 2008b), uranium (Moll, 2008a) and neptunium (Moll, 2010). Interactions between SRB and curium have also been identified (Moll, 2004). Finally, it has been found that micro-organisms can sorb radionuclides on their cell surfaces (Pedersen and Albinsson, 1991), thereby facilitating mobilisation.

H_2 is readily used by many different phylogenetic traits of anaerobic micro-organisms, such as SRB (Barton, 2009), acetogens (Drake, 2002) and methanogens (Ferry, 1992). Of particular interest is the H_2 produced during the anaerobic corrosion of iron components in a repository, for example, the rock bolts installed to secure the tunnel from falling rocks. Such corrosion, evolving H_2, may lead to the local production of large amounts of sulphide (Pedersen, 2012a, 2012b, 2013). In HLRW repositories with sulphate-rich groundwater there is a strong potential for microbial sulphate reduction to sulphide via two metabolic processes. First, in addition to the naturally occurring H_2 in groundwater, iron in water-filled deep underground repository construction is bound to corrode anaerobically with the concomitant production of H_2 (Reardon, 1995), which may induce SRB growth and sulphide production. Second, in the case where sulphate-rich groundwater mixes with deep methane-rich groundwater SRB growth and activity can be induced (Pedersen, 2013). Regarding the anaerobic metabolism of methane, the situation is obscure. Although anaerobic oxidation of methane (AOM) with sulphate as final electron acceptor is a well-documented process, detailed information about the metabolic pathways of AOM awaits successful cultures (Knittel, 2009). The electron donor is likely to be methane, but more research is required before conclusions can be drawn regarding the detailed nature and extent of AOM processes in HLRW environments.

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Iron/argillite interactions in radioactive waste disposal context: Oxidising transient and bacterial activities influence

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Introduction

The design of a high-level radioactive waste (HLW) disposal facility developed by Andra (2005) in France involves emplacing metallic materials (containers, overpacks, liner) into a geological argillaceous formation. During the operational phase, ventilation of handling drifts will keep oxidising conditions at the front of disposal tunnels. Therefore, an oxidising transient may take place in parts of these tunnels in the post-closure phase possibly over several years. During this transient period, the environment of the disposal cell will evolve towards reducing and saturated conditions close to the equilibrium state of the original underground argillaceous formation. Moreover, high temperature conditions above 50°C may be encountered in this environment over a few hundred years.

Uniform corrosion represents the main type of degradation of metallic materials for the long term. The oxidising transient will be characterised by high corrosion rates (e.g. localised corrosion) due to the presence of oxygen whereas during the following anoxic stage, the main alteration factor will originate from the pore water associated with lower corrosion rates (Féron, 2008). In any case, metallic materials corrosion will lead to the release of aqueous iron, which may induce alteration of the favourable confining properties of the clayey materials. In this context, reactive pathways related to the metal corrosion under oxidising conditions and then followed by reducing conditions remain to be further understood (evolution of pH, redox and influence of temperature). Furthermore, some other significant issues remain open, in particular the dissolution/precipitation processes, the argillite perturbation extent and the effects of these transformations on the confining properties of materials.

The presence of micro-organisms in deep argillaceous environment (Poulain, 2006; Stroes-Gascoyne, 2007; Urios, 2012) and the introduction of new bacterial species in the repository during the operational phase raise the question of their survival under real disposal conditions. Indeed, the nature, the quantity of nutrients and the environmental conditions (space, temperature, water, radioactivity and pressure) are key parameters for bacterial development. Even though disposal conditions may be not favourable during a part of the thermal transient characterised by high temperature, irradiation conditions and heterogeneous water saturation, bacterial activity may resume when environmental conditions become more suitable (Motamedi, 1996). Moreover, argillite cracks and residual voids between the waste packages and the liner create additional space for bacterial development. Concerning the nutrient content, significant amounts of hydrogen (an energetic substrate for bacteria) produced by anoxic corrosion of metallic materials are expected, which will favour the development of hydrogenotrophic bacteria (Libert, 2011). Furthermore, it is widely accepted that micro-organisms may locally affect the corrosion processes and the corrosion rates due to their influence on the water composition, pH and redox potential of the metal/environment interface (Beech, 2004). More specifically, sulphate-reducing bacteria (SRB) may produce ferrous sulphide, a corrosive product that may lead to significant pits on steel surface. Also, under anaerobic conditions, the iron-reducing bacteria (IRB) can reduce Fe(III) from iron oxides composing passive layers, which may impact corrosion by re-exposing the metal surfaces to corrosion. Therefore, the survival of bacteria cannot be excluded and their impact on corrosion phenomena must be investigated.

In this context, this paper focuses on two studies regarding iron/argillite interactions. The first one addresses these interactions under oxidising and reducing conditions, while the second one tackles bacteria effects on corrosion in conditions that may prevail in a repository. These studies are both based on laboratory and *in situ* experiments. Iron and carbon steel have been chosen as typical of metallic components, and the Tournemire Toarcian argillite (Charpentier, 2003) has been selected as a representative argillaceous rock. These studies aim at acquiring additional data which will allow to strengthen the existing models, in particular regarding their space and time extrapolation.

Scientific approach and main results

A ten-year *in situ* experiment was conducted at the IRSN's Tournemire underground laboratory to understand interaction processes between these two materials, steel and argillite, under disposal conditions. This *in situ* test consisted in emplacing five carbon steel discs within crushed and compacted argillite in the CR6 horizontal borehole drilled from the main tunnel (Dauzères, 213; Gaudin, 2009)). After ten years of interaction, samples were extracted by over-coring (diameter 250 mm) and preserved from oxidation. A complete characterisation stage was carried out to identify the physical-chemical evolution of this steel/argillite interface.

Investigation of steel/argillite composition evidences: *i)* a precipitation of corrosion products with an alteration thickness ranging from 50 μm to 300 μm; *ii)* a heterogeneous diffusion of the released iron into the argillite with an extent estimated around 5 mm (Figure 1). This difference in perturbation thickness is linked with the occurrence of several cracks in the argillite. The occurrence of such ccracks may enhance local corrosion by creating preferential pathways for aggressive ionic species, but it also acts as preferential pathways for released iron and induces heterogeneous perturbations of the argillite. Corrosion products are mainly composed of hematite, magnetite and siderite. Corrosion rates (estimated from weight loss) decrease between two and six years, which suggests that the oxidising transient lasted more than two years. Around the interface, an increase in sulphur concentration is detected, which may be related with the presence of SRB (Figure 2). Regarding the argillite perturbation, a decrease in calcium concentration is observed. This might be interpreted as a dissolution of calcite and/or as a decrease of the Ca-smectite fraction in the illite/smectite mixed layer due to local changes in chemical conditions of the environment.

An important porosity variation is evidenced at the argillite/carbon steel interface. The steel corrosion zone shows large porosity increases (up to 30%) within the corrosion products. On the opposite, the high porosity of the crushed argillite (45%) falls down to 20% at the interface. These results indicate a partial clogging of the clayey material porosity resulting from the iron perturbation.

Moreover, to assess the influence of bacteria on iron/argillite interactions, two percolation experiments were carried out at 60°C (temperature representative of those encountered close to the disposal cell over a few hundred years), with iron (powder and rod) and Toarcian argillite (Matray, 2007) involving an artificial crack. Both experiments lasted 13 months, one under biotic conditions (with SRB and IRB) and the other under abiotic conditions (no bacteria) (Chautard, 2012). In the first one, two hydrogenotrophic bacteria species were introduced, namely a SRB and an IRB. The second experiment was carried out as a control experiment without bacteria in order to highlight specific processes induced by the presence of bacteria. Solution chemistry (major anions and cations concentration) was monitored over time, the characterisation of the solid phases (SEM, TEM, μRaman, μXRD) was carried out at the opening of cells (anoxic conditions were preserved as much as possible in all subsequent analyses).

Figure 1: Carbon steel/argillite interface; iron diffusion into the argillite

(a) Sample photography; (b) 3-D XR microtomography section highlighting iron gulfs in argillite

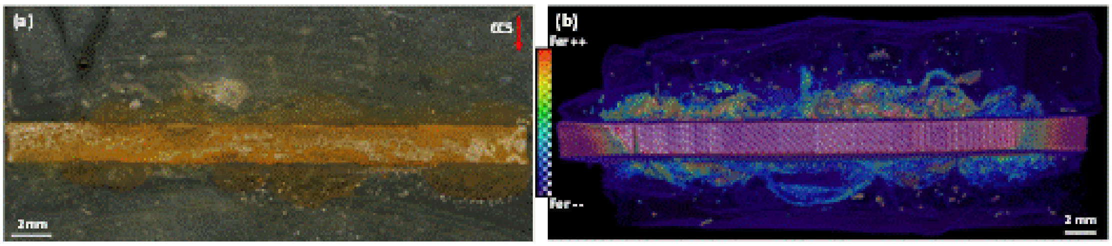
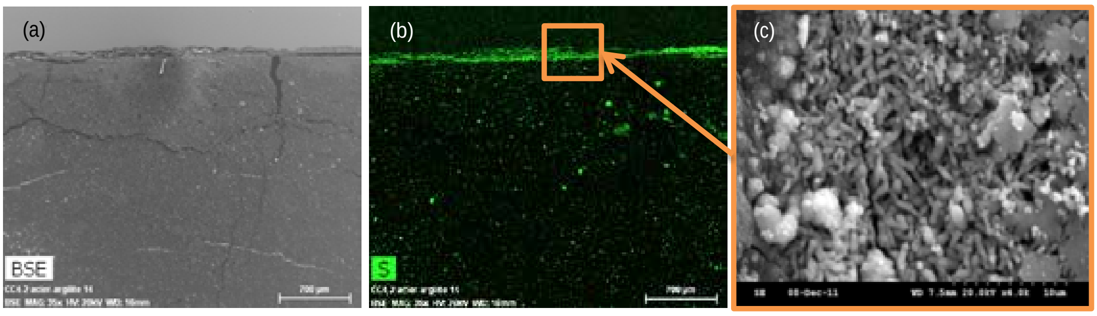


Figure 2: Scanning electron microscopy (SEM) image by backscattering electrons (BSE) of the iron argillite corroded interface

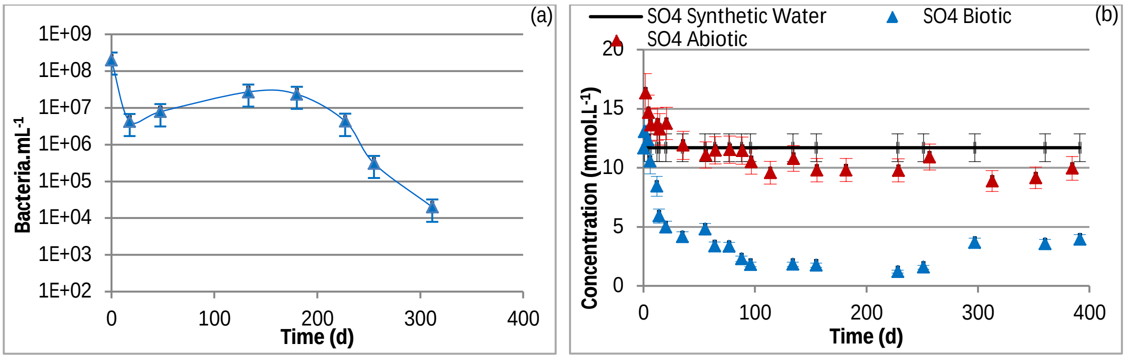
(a) EDX chemical mapping of the same area for sulphur
(b) Bacteria biofilm in argillite near the sulphur enrichment at the interface (c)



After 13 months of experiment, the characterisation of the microbial diversity at the end of the test confirmed that bacteria were still present in the biotic cell. The bacteria population monitoring (epifluorescence method) evidences a decrease of their number after 180 days, although their presence remains significant [Figure 3(a)]. The decrease of sulphate concentration is also observed in the biotic cell, which confirms their activities [Figure 3(b)] under the experiment conditions (nutrient-poor and small available space). Bacteria survival under similar conditions has also been observed by El Hajj, *et al.* (2010).

The reactivity of the biotic experiment is evidenced by the specific evolution of the water composition at the bottom of the cell (pH, Ca, K, Mg) (Chautard, 2012). Around iron powder, observations on both experiments highlighted the formation of corrosion products, namely magnetite and chukanovite. Moreover, iron migration (10 μm) has been evidenced by SEM-EDX, the larger extents being associated with micro-cracks formed during the experiment. A decrease in calcium concentration all along the initial fracture and the micro-cracks is also observed and might be interpreted as a perturbation induced by the percolating fluid. Such a calcium depletion has already been observed in the *in situ* experiment. At the present characterisation stage, no significant difference between both experiments is evidenced thus far, except the local and heterogeneous sulphur increase

Figure 3: Evolution of bacteria population (a) and sulphate concentration (b)



with bacteria observed around the iron rod. This may be explained by low bacterial activity due to harmful conditions of the experiment (small available space and nutrient-poor). Additional characterisation data will help to further investigate these processes, including the potential impact of bacteria on iron (powder and rod)/argillite interactions.

Conclusion

The oxidising transient has been shown to be a key parameter controlling the system evolution during several years under the tested conditions. This phenomenon and its kinetics should be further considered and characterised to better assess its impact on reactive pathways.

Bacteria survival in an experiment representative of HLW disposal conditions has been confirmed, though their population decreases due to the environment conditions harmful to bacteria. After 13 months of experiment, the reactivity of the biotic experiment is evidenced by the specific evolution of the water composition (pH, Ca, SO_4), although no significant solid phases differences are evidenced so far, except a local sulphur increase observed around the iron rod. As bacteria activity may be an important parameter in the geochemical evolution of the iron/argillite system, further investigations must be conducted to better assess their effects on corrosion processes.

Two IRSN projects, namely OXITRAN and BIOFILM, are underway to study bacterial and redox transient effects on corrosion processes under HLW conditions.

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