POSIVA 2010-02

Interim Summary Report of the Safety Case 2009

Posiva Oy

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INTERIM SUMMARY REPORT OF THE SAFETY CASE 2009

Tiivistelmä – Abstract

Following the guidelines set forth by the Ministry of Trade and Industry (now Ministry of Employment and Economy), Posiva is preparing to submit a construction license application for the final disposal spent nuclear fuel at the Olkiluoto site, Finland, by the end of the year 2012. Disposal will take place in a geological repository implemented according to the KBS-3 method. The long-term safety section supporting the license application will be based on a safety case that, according to the internationally adopted definition, will be a compilation of the evidence, analyses and arguments that quantify and substantiate the safety and the level of expert confidence in the safety of the planned repository. The present Interim Summary Report represents a major contribution to the development of this safety case. The report has been compiled in accordance with Posiva's current plan for preparing this safety case. A full safety case for the KBS-3V variant will be developed to support the Preliminary Safety Assessment Report (PSAR) in 2012.

The report outlines the current design and safety concept for the planned repository. It summarises the approach used to formulate scenarios for the evolution of the disposal system over time, describes these scenarios and presents the main models and computer codes used to analyse them. It also discusses compliance with Finnish regulatory requirements for long-term safety of a geological repository and gives the main evidence, arguments and analyses that lead to confidence, on the part of Posiva, in the long-term safety of the planned repository.

Current understanding of the evolution of the disposal system indicates that, except a few unlikely circumstances affecting a small number of canisters, spent fuel will remain isolated, and the radionuclides contained within the canisters, for hundreds of thousands of years or more, in accordance with the base scenario. Confidence in this base scenario derives, in the first place, from the intrinsic properties of the main components of the repository and from the understanding of their evolution gained from extensive site- and concept-specific field, laboratory and modelling studies and from studies of natural and anthropogenic analogues.

For any canisters that fail over this time window, the low radionuclide calculated release rates to the biosphere and resultant annual effective doses to humans and absorbed dose rates to other species of flora and fauna imply that any radiological consequences of these releases will be negligible. Furthermore, the calculation results indicate that, in general, differences in the geometry and transport paths considered in the analyses of the KBS-3V and KBS-3H design variants have only a minor impact on calculated releases and doses.

Work carried out to date indicates that a geological repository for the final disposal of spent fuel, implemented as planned at the Olkiluoto site, will conform to Finnish regulatory requirements and provide an adequate level of long-term safety. This conclusion is based on the findings of safety assessments, the systematic treatment of uncertainty in these assessments and the quality measures that have been applied in the development and application of models, data and computer codes. Plans are in place to manage remaining safety-related issues and uncertainties, as given in the report TKS-2009. In implementing TKS-2009, quality assurance measures will be applied in the various production steps of the safety case, including and tests and experiments to demonstrate the feasibility and quality of technical solutions. In this way, a comprehensive safety case will be developed to support the licensing process.

Avainsanat - Keywords

safety case, safety assessment, KBS-3V, KBS-3H, Olkiluoto

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TURVALLISUUSPERUSTELUN YHTEENVETORAPORTIN ALUSTAVA VERSIO 2009

Tiivistelmä – Abstract

Kauppa- ja teollisuusministeriön vuonna 2003 vahvistaman aikataulun mukaisesti Posiva on valmistautumassa käytetyn ydinpolttoaineen loppusijoituslaitoksen rakentamislupahakemuksen jättämiseen vuoden 2012 lopulla. Loppusijoituksen pitkäaikaisturvallisuus käsitellään lupahakemuksessa ns. turvallisuusperusteluna (engl. *safety case*), jolla kansainvälisesti omaksutun määritelmän mukaisesti tarkoitetaan kaikkea sitä teknis-tieteellistä aineistoa, analyysejä, havaintoja, kokeita, testejä ja muita todisteita, joilla perustellaan loppusijoituksen turvallisuus ja turvallisuudesta tehtyjen arvioiden luotettavuus. Vuonna 2008 Posiva esitti päivitetyn suunnitelman tuvallisuusperustelun muodostavasta aineistosta ja sen laatimisesta. Tämä raportti on alustava versio Posiva Oy:n Olkiluotoon suunnitellusta KBS-3 ratkaisuun ja geologiseen loppusijoitukseen perustuvasta käytetyn polttoaineen loppusijoituslaitoksen turvallisuusperustelun yhteenvetoraportista. Täydellinen turvallisuusperustelu, mukaan lukien päivitetty biosfääriarviointi, laaditaan alustavaa turvallisuusselostetta (PSAR) varten 2012.

Tässä raportissa esitellään loppusijoituksen tämän hetkinen tekninen suunnitelma ja turvallisuuskonsepti, sekä yhteenveto loppusijoitusjärjestelmän kehityskulun tarkastelussa käytettävien skenaarioiden muodostamisesta, jossa kuvataan eri skenaariot ja esitellään tärkeimmät analysoinnissa käytetyt mallit ja tietokonekoodit. Loppusijoituksen pitkäaikaisturvallisuutta arvioidaan myös viranomaisvaatimusten täyttymisen osalta. Raportissa esitellään myös teknis-tieteellistä aineistoa, analyysejä, havaintoja, kokeita, testejä ja muita todisteita, joilla osoitetaan loppusijoituksen pitkäaikaisturvallisuus.

Tutkimukset tähän päivään asti osoittavat muutamaa epätodennäköistä tapahtumaa lukuun ottamatta, jotka vaikuttavat vain muutamaan kapseliin, että käytetty polttoaine säilyy koskemattomana kapseleissa useiden satojen tuhansien vuosien ajan perusskenaarion mukaisesti. Luottamus perusskenaarioon perustuu ennen kaikkea loppusijoituksen ensisijaisten vapautumisesteiden ominaisuuksiin ja niiden kehityskulun ymmärtämiseen, joka on saavutettu laajamittaisilla paikka- ja konseptikohtaisella kenttä-, laboratorio- ja mallinnustutkimuksilla, sekä luonnon- ja antropogeenisilla analogioilla.

Yksittäisen kapselin rikkoutumisesta ensimmäisten kymmenentuhannen vuoden arviointiajanjaksolla aiheutuvat säteilyannokset ja niiden vaikutukset ovat vuositasolla merkityksettömän pieniä sekä ihmisten että muiden eliölajien osalta aiheutuva radionuklidien vapautuminen biosfääriin on vuositasolla annoksiltaan merkityksetöntä. Lisäksi voidaan todeta, että analyyseissä käytetyt geometriset ja vapautumisreittejä koskevat muuttujat eivät vaikuta merkittävästi vapautumis- ja säteilyannostuloksiin. Tässä suhteessa KBS-3H ja KBS-3V vaihtoehtojen välillä ei ole merkittävää eroa.

Johtopäätöksenä voidaan todeta, että käytetyn polttoaineen geologinen loppusijoitus, toteutettuna kuten on suunniteltu Olkiluodossa, vastaa viranomaisvaatimuksia ja takaa riittävän pitkäaikaisturvallisuuden. Edellä esitetty johtopäätös perustuu tähän mennessä tehtyihin turvallisuusarvioihin ja niissä systemaattiseen epävarmuustekijöiden tarkasteluun, sekä mallien kehittämiseen, mallinnukseen, tietokantoihin ja tietokonekoodeihin sovellettuun laadunvarmistukseen. Vielä avoinna olevat turvallisuuteen liittyvät epävarmuudet on eritelty TKS-2009 raportissa, jossa myös suunnitelmat niiden ratkaisemiseksi on esitetty. TKS-2009 suunnitelmien toteutuksessa laadunvarmistusmittareita käytetään useilla tasoilla; näihin sisältyy myös teknisten ratkaisujen toteutettavuuden kokeellinen testaus ja laatu. Näitä suunnitelmia noudattamalla voidaan tuottaa kattava turvallisuusperustelu, jota voidaan käyttää tukena lopullisessa loppusijoituslaitosta koskevassa lupamenettelyssä.

Avainsanat - Keywords
turvallisuusperustelu, turvallisuusanalyysi, KBS-3V, KBS-3H, Olkiluoto

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Executive summary

The present report is an interim Summary Report outlining Posiva's current safety case for the final disposal of Finnish spent fuel in a geological repository at the Olkiluoto site, developed according to the KBS-3 method, based on safety assessments carried out to date. It forms part of a safety case report portfolio, the main reports of which are shown in Fig. 1.

A more complete safety assessment will be carried out, and the safety case updated, to support the Preliminary Safety Assessment Report (PSAR) in 2012.

Currently, two variants of the KBS-3 method are under consideration - KBS-3V and KBS-3H - as illustrated in Figure 2. Ongoing safety studies for a KBS-3V repository at Olkiluoto have resulted in production of a number of reports, including, most recently, Posiva's 2009 safety analysis of a KBS-3V repository. This includes a radionuclide transport study, RNT-2008, which assesses the potential magnitude of any release of radionuclides from the repository to the surface environment, and a biosphere assessment BSA-2009, which includes modelling of the transport of these radionuclides within the surface environment and an assessment of the possible radiological consequences to humans and other biota. Posiva and SKB have also conducted a joint Research, Demonstration and Development programme in 2002-2007 with the overall

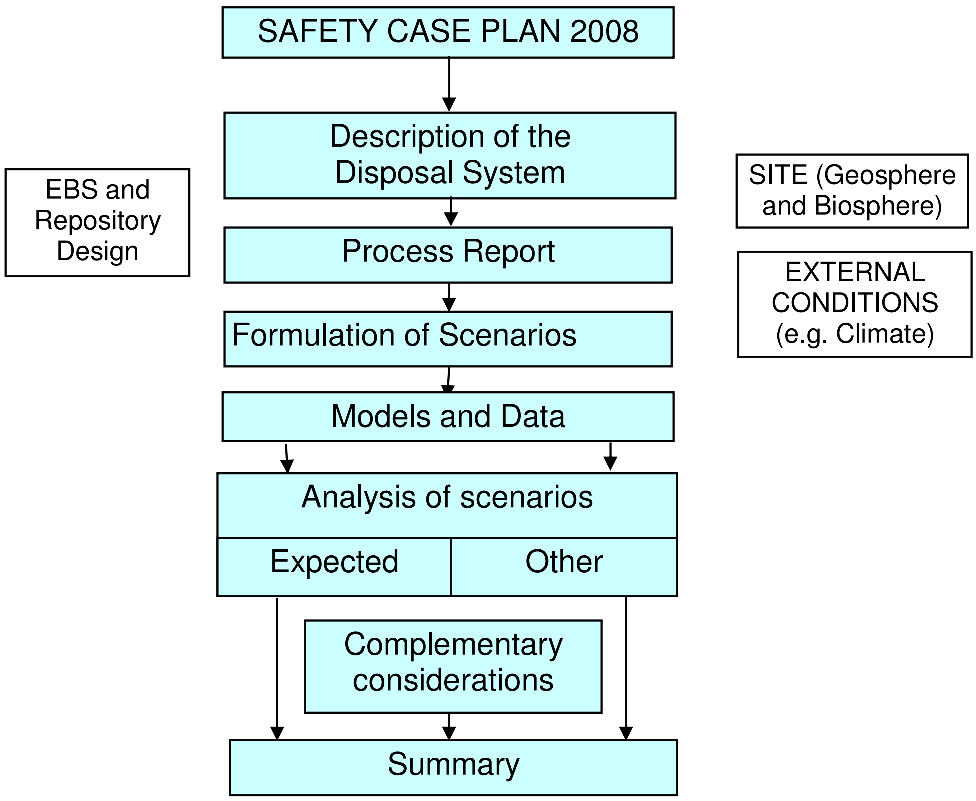

Figure 1. *Main reports of the safety case portfolio (in blue) and the main input from supporting technical and scientific activities (in white). EBS: engineered barrier system.*

aim of establishing whether KBS-3H represents a feasible alternative to KBS-3V. A summary report describing the safety assessment of a KBS-3H repository at Olkiluoto was published in 2008. The findings of the KBS-3V and KBS-3H safety studies are summarised in the present report. A full safety case for the KBS-3V variant of the KBS-3 method will be carried out to support the Preliminary Safety Assessment Report (PSAR) in 2012. The KBS-3H alternative will be analysed using a common methodology and the cases studied will differ only to the extent that this is required by the different designs. A full safety case for the KBS-3H alternative will be carried out after PSAR. According to current planning, the decision of which variant finally to implement can be made in the run up to the construction of the underground facilities in 2015-2016.

Current design and safety concept

In both variants of the KBS-3 method, spent fuel encapsulated in canisters is emplaced deep underground in a geological repository. The canister consists of a cast iron insert enclosed entirely within a copper overpack. In KBS-3V, the canisters are emplaced vertically in individual deposition holes excavated in the floors of deposition tunnels. In KBS-3H, several canisters are emplaced horizontally in a system of 100-300 m long deposition drifts. In both variants, the canisters are surrounded by a swelling clay buffer material that separates them from the bedrock and, in the case of KBS-3H, also separates the canisters one from another along the deposition drifts. In current KBS-3H designs, each canister is pre-packaged in an assembly, called a supercontainer, before emplacement in the deposition drifts. In the current design, the supercontainer consists of a perforated steel shell cylinder containing the canister surrounded by a layer of bentonite, although alternative materials for the shell are also being studied. The KBS-3V deposition tunnels are backfilled with a swelling material of low permeability. The transport tunnels and other underground openings in both variants are also backfilled with a low permeability material.

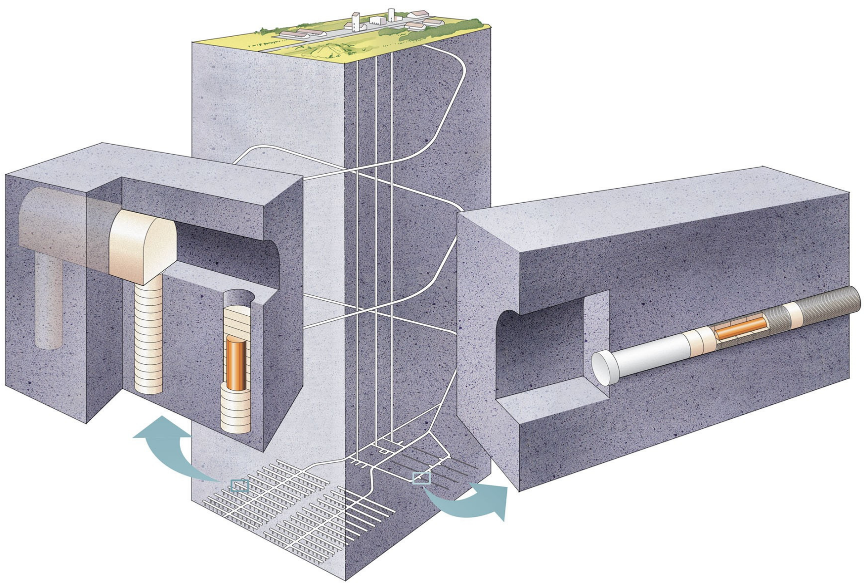

__Figure 2.__ The KBS-3V (left) and KBS-3H (right) alternative realisations of the KBS-3 spent fuel disposal method.

According to the safety concept for the KBS-3 method, long-term safety is achieved by isolating the spent fuel deep underground, and containing its radionuclides by a system of multiple barriers, both engineered and natural, which ensure that no single harmful event or deficiency of the system may endanger the ability of the system to provide safety (Fig. 3). The main long-term safety barriers are:

- the canister;
- the buffer;
- the backfill in the deposition tunnel (KBS-3V); and
- the bedrock,

although other, auxiliary components, can also play a long-term safety role.

In the KBS-3 method, containment of the radionuclides associated with the spent fuel is provided first and foremost by encapsulating the fuel in water-tight and gas-tight sealed canisters. Providing a prolonged period of complete containment of radionuclides is the main safety function of the canister. Fig. 3 shows as red pillars and blocks key features of the system supporting the operation of this safety function. The host rock isolates the spent fuel from the biosphere and normal habitat of humans and other biota and limits the possibility of human intrusion. Other safety functions of the host rock, the buffer and the KBS-3V deposition tunnel backfill relate to the protection of the canisters, and to the provision of adequate levels of safety in the event of canister failure. Fig. 3 shows as green pillars and blocks secondary features of the system that ensure the retention of radionuclides and the limitation and retardation of radionuclide releases from a failed canister.

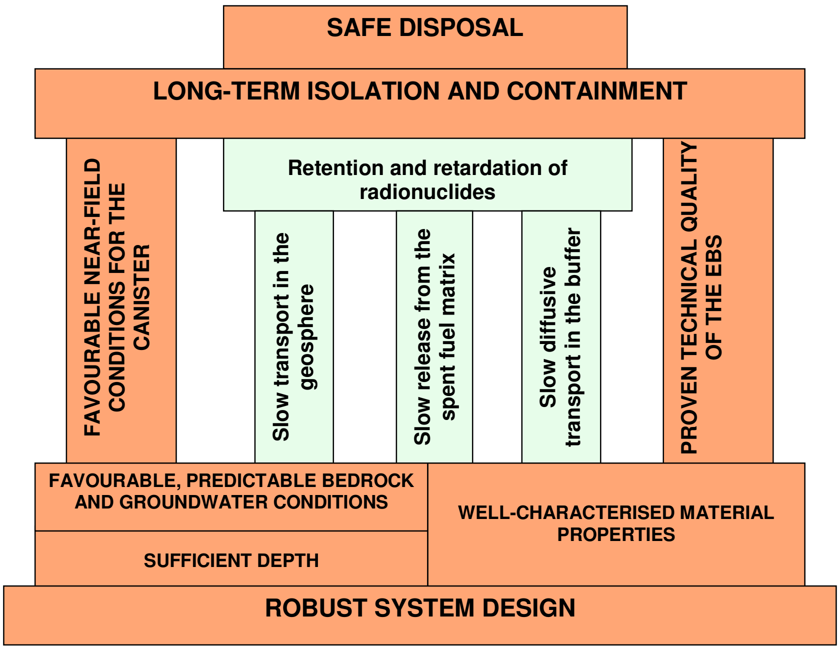

Figure 3. *Outline of safety concept for a KBS-3 type repository for spent fuel in a crystalline bedrock.*

Performance targets have been defined for critical parameters determining the long-term performance of the engineered barriers, and these will be further developed and applied in the in support the Preliminary Safety Assessment Report (PSAR) in 2012. Target properties contributing to the performance of the engineered barriers and retention of radionuclides are defined for the host rock as part of a set of rock suitability criteria (RSC). If the performance targets are achieved, and target properties are present, then the repository barriers are expected to fulfil their respective safety functions. If plausible situations can be identified where the performance targets or target properties are not achieved, then the consequences of loss or degraded performance of the corresponding repository barrier or barriers must be evaluated as part of the safety assessment.

Assessment basis and methodology

Theoretical or conceptual understanding of relevant features, events and processes (FEPs) and their interactions forms the basis for assessing the long-term safety of the proposed repository and producing a safety case. The current understanding of FEPs is documented, to a large extent, in various versions of the Process Report, which is a main report in the current safety case portfolio (Fig. 1). An important consideration in compiling the Process Report is completeness. Thus, the latest version of the Process Report for the KBS-3V variant was cross-checked for completeness against, for example, the NEA international FEP database.

Features that favour a long canister lifetime are its mechanical strength and its corrosion resistance. The mechanical strength of the canister is provided mainly by its cast iron insert. The minimum collapse pressure of the canister is significantly higher than the loads to which the canister is expected to be subjected over time. The cast iron insert is protected from corrosive agents in the surrounding water by the copper overpack. If the buffer performs as expected, several hundreds of thousand of years or more will elapse before the overpack corrodes significantly. Key features of the clay are its low hydraulic conductivity, its swelling pressure, its chemical buffering capacity, its sorption capacity and its plasticity. For example, because of the low hydraulic conductivity of the buffer once saturated, migration of corrosive agents from the groundwater to the canister surface will occur only slowly, predominantly by molecular diffusion. Favourable features of the bedrock at the Olkiluoto site include its low seismic activity, its lack of exploitable resources that might lead to inadvertent human intrusion, its sparse fracturing and the low groundwater flow rates at repository depth and geochemical conditions that are broadly favourable to backfill, buffer and canister longevity and performance. Although the rock is spatially variable, rock suitability criteria (RSC) will be used to ensure that deposition holes or tunnels are located at positions that provide a protective environment for the canisters and buffer and backfill and limit the radionuclide transport in the event of canister failure.

A range of processes will operate that will severely limit the release of radionuclides to the surface environment in the event of canister failure. For example, the fuel matrix incorporates the majority of the radionuclide inventory and is resistant to degradation by water entering the canister in the expected chemical environment. The chemical environment within and around the repository, and especially the reducing conditions, will favour low solubility and high sorption of many safety-relevant radionuclides. Groundwater flowing in fractures will convey released radionuclides through the bedrock. However, the RSC will ensure low groundwater flow in the vicinity of the canisters. Furthermore, many radionuclides will be strongly retarded by matrix diffusion and sorption on rock matrix pore surfaces, and will decay to insignificant concentrations during transport. The most important transport processes are described further in Chapter 4, and quantitatively evaluated in Chapters 7 and 8.

In spite of the favourable features outlined above, features, events and processes that could potentially lead to canister failure, or degrade the capacity of the repository to limit radionuclide transport in the event of canister failure, cannot be excluded. These various uncertain features, events and processes are taken into account in the definition and formulation of scenarios

The classification of scenarios adopted by Posiva for the purposes of the recent safety assessments is shown in Fig. 4. This will updated for the PSAR 2012 based on guidance in YVL E.5.

Climatic scenarios provide the framework within which the internal evolution of the disposal system can be described. The disposal system is taken to comprise the repository system, i.e. the system of engineered barriers and the surrounding bedrock, plus the overlying surface environment.

In the safety analyses described in the present report, the **base scenario** includes all lines of evolution of the disposal system giving no release of radionuclides. The surface environment is not expected to affect the integrity of the repository system during the

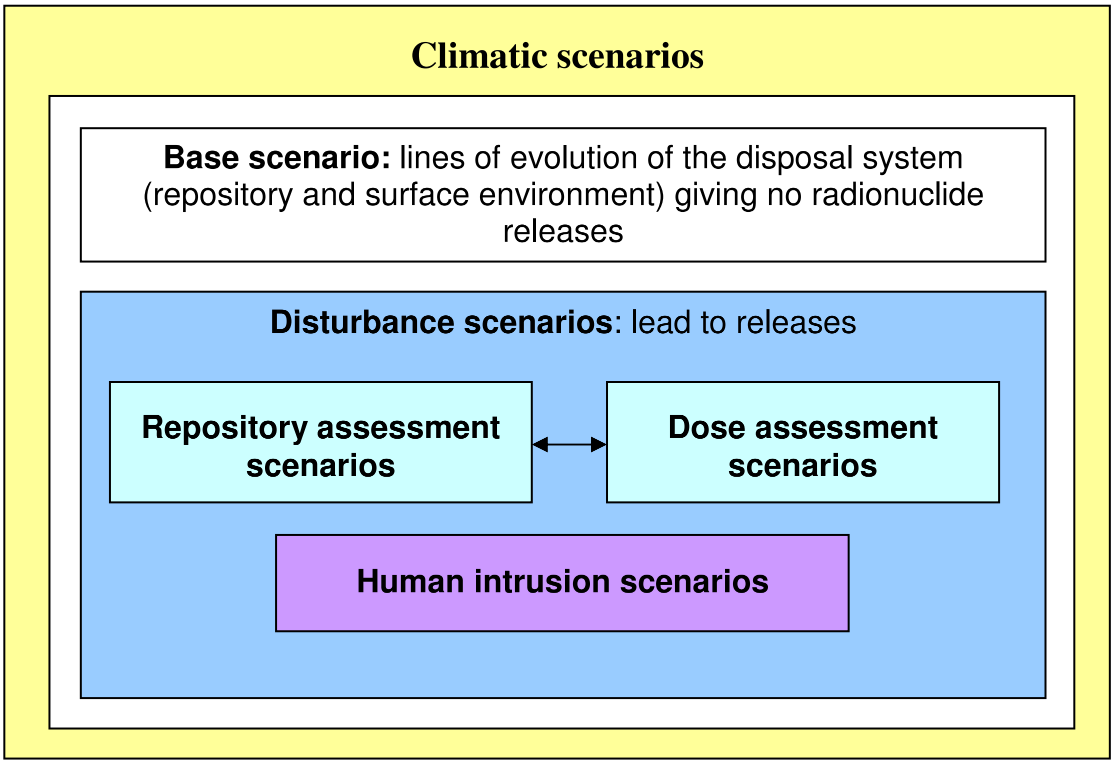

Figure 4. *Scenario classification in the safety case.*

first several thousands of years after repository closure for any plausible climatic scenario. Thus, all lines of evolution of the surface environment during the first several thousands of years are included in the base scenario (this is also the time window for which a quantitative dose assessment is required by Finnish regulations).

Disturbance scenarios include those lines of evolution of the disposal system that include radionuclide release and hence to the possibility of exposure of humans and other biota to ionising radiation. They are developed by combining repository assessment scenarios and dose assessment scenarios, and also include human intrusion scenarios. These categories of scenarios are each described separately, below.

Repository assessment scenarios are developed for lines of evolution of the repository system leading to canister failure and radionuclide release. These generally have a low probability of occurrence, although in some cases the probabilities are not yet well defined.

Dose assessment scenarios are scenarios describing the potential fate of radionuclides in the surface environment. They include lines of evolution of the surface environment, which also form part of the base scenario, and lines of evolution for how humans and other biota inhabit and use the surface environment during the time window for quantitative dose assessment (at least several millennia), taking regulatory guidelines into account. The evolution of the surface environment may, to some extent, affect conditions deep underground and thus may have some impact on the ways in which repository assessment scenarios are modelled.

In addition to the scenarios mentioned above, the consideration of unintentional disturbance of, or intrusion into, the repository by humans subsequent to repository closure is a requirement in Finnish regulations. Uncertainties in the evolution of human society and of the state-of-the-art in science and technology are such that estimates of both probability and consequences of **human intrusion scenarios** must be based on "stylised assumptions" that cannot be fully substantiated or evaluated in respect to conservatism of radiological consequence estimates. They are thus considered as a class of scenarios separate from repository assessment scenarios.

The fate of radionuclides arising from disturbance scenarios is analysed using quantitative models. The general modelling as approach is illustrated in Fig. 5.

Conservatively[1], it is assumed that radionuclides may migrate from the repository near field (the engineered barriers and the disturbed part of the bedrock that surrounds them) to the geosphere (the remainder of the bedrock), but not vice-versa. Similarly, it is conservatively assumed that radionuclides may migrate from the geosphere to the biosphere, but again not vice-versa.

Groundwater flow modelling and surface and near-surface hydrological modelling, supported by the descriptions of the host rock and surface environment and their

[1] A conservative assumption in the present context is one that is expected to overestimate radiological consequences.

evolution, provide key inputs to the principal radionuclide release and transport models shown as white boxes in Fig. 5.

Posiva's approach to assessing the radiological consequences of geosphere releases comprises the development of a description of present conditions and transport and development processes in the surface systems and, on the basis of this, (i), predicting - or forecasting - the development of the topography, surface hydrology, flora and fauna, by means of terrain and ecosystems development modelling (TESM), (ii), transport modelling using landscape models with different configurations, and (iii), assessing potential radiological consequences of radionuclide releases to humans and other biota.

Each scenario comprises lines of evolution that are analysed or assessed with calculation cases, taking into account model and parameter uncertainties. The assessment models are generally simplified representations of the actual disposal system. Simplifications are generally justified on the grounds of conservatism or insignificant consequences. In applying the models, uncertainties in input data are

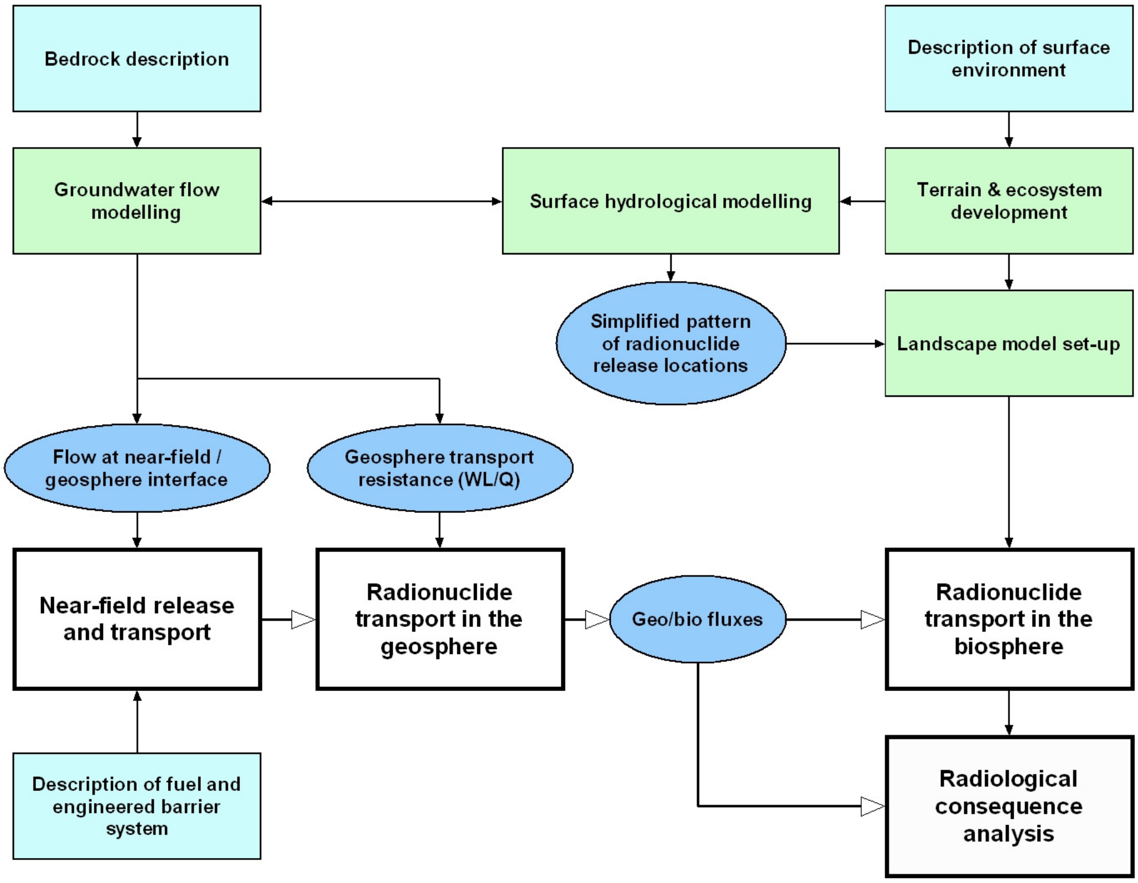

Figure 5. *Models and information flows. Radionuclide release and transport models and consequence analysis are shown in white boxes. System descriptions are shown in light blue boxes, key supporting models in green boxes and their principal outputs in dark blue ovals.*

generally treated either by cautious parameter value selection, or by exploring the consequences of alternative parameter values.

In the 2009 safety assessment of a KBS-3V repository and in the KBS-3H safety assessment, as in earlier Finnish safety assessments, a “deterministic” modelling approach to assessing the impact of specific model and parameter uncertainties has been adopted, involving:

- defining and modelling a base calculation case for the whole analysis (KBS-3V) or separate base calculation cases for each identified canister failure mode (KBS-3H);
- identifying alternative conceptual assumptions and parameter values consistent with current scientific understanding;
- defining and modelling variant calculation cases (i.e. sensitivity cases, “what if” cases, and complementary cases) that incorporate these alternatives either individually or in combination.

Disposal system evolution in the base scenario

The starting point for the description of disposal system evolution in the base scenario, and also in other scenarios, is a description of its initial state. The conditions for the surface environment predicted for the year 2020, upon the emplacement of the first canister, define the initial state of the biosphere, and are based on forecasts from the terrain and ecosystems development modelling. The initial state of the engineered barriers is obtained largely from the design specifications of the repository, including allowed tolerances for uncertainties and deviations, as ensured by quality control measures. The initial state of the near-field rock and, to some extent, the broader geosphere, is determined - or at least constrained - by the rock suitability criteria (RSC - still under development) that are applied in locating suitable rock volumes for repository construction and for canister deposition. The initial state of the geosphere will differ from the undisturbed conditions in the bedrock due to the effects of repository construction (including the construction of the underground rock characterisation facility ONKALO). These effects are quantified by means of rock mechanics, thermal and groundwater flow modelling, including modelling of the transport of dissolved chemical species.

The repository system will evolve from its initial state through an early, transient phase towards a target state, in which the safety functions are fulfilled. Once the target state has been reached, the key safety-relevant physical and chemical characteristics (e.g. temperature, buffer density and swelling pressure) are subject to much slower changes than in the transient phase. The main transient processes occurring within and around a deposition hole in a KBS-3V repository are illustrated in Fig. 6. Two key transient processes - heat transfer from the spent fuel to the surrounding rock and saturation of the repository with groundwater - may take up to several thousands of years, and even longer in the case of saturation of the tightest parts of the rock, and this may be taken as the rough duration of the early evolution period.

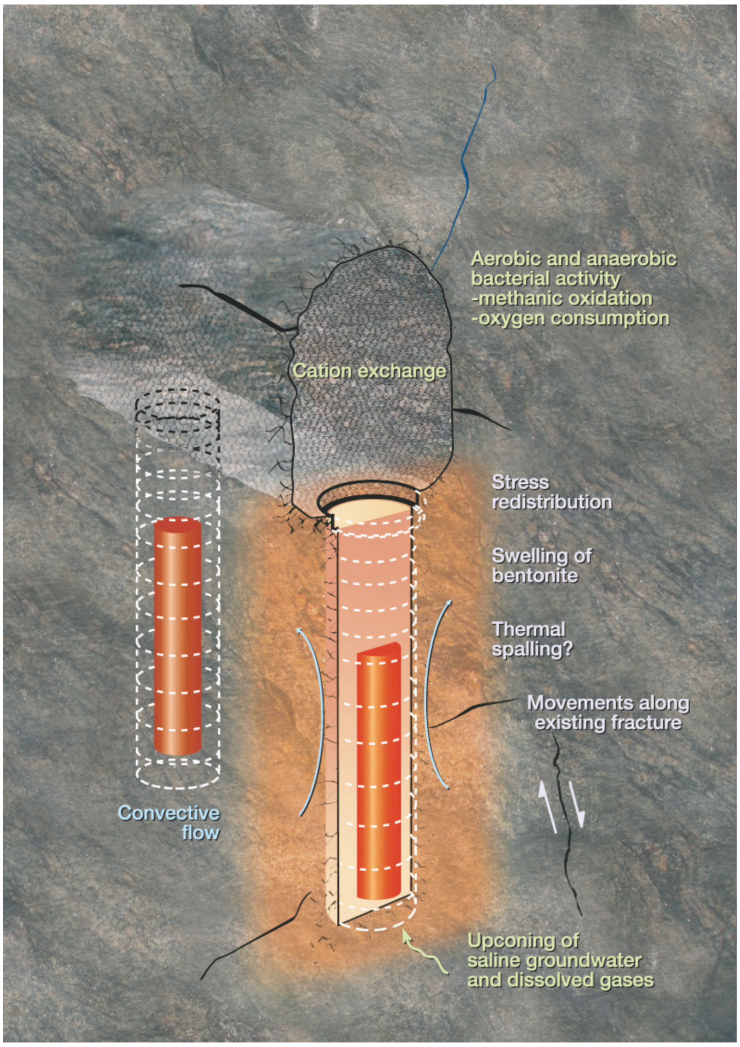

Figure 6. *Main processes in the deposition hole during the early evolution of a KBS-3V repository. Geologic features (e.g. rock stresses and fractures) are not to scale and are exaggerated for clarity.*

Largely similar processes occur within a KBS-3H deposition drift, although there are some processes that are specific to each variant, and it is in the early, transient phase that most of the significant differences in evolution between the KBS-3V and KBS-3H variants arise. For example, particularly in tight KBS-3H drift sections, the gas generated by the corrosion of steel components external to the canister (principally the supercontainer shell in the current reference design) may accumulate at the buffer / rock interface, resulting in a prolonged period during which inflow of water from the surrounding rock will be limited, which will delay saturation of the buffer.

In the current base scenario, the copper coverage of the canisters remains intact during the period of early evolution and beyond. Furthermore, in this scenario, appropriate quality control procedures during encapsulation and emplacement ensure that the canisters have no initial penetrating defects. As a consequence, the radionuclides associated with spent fuel are totally contained within the canisters in this scenario.

During the period of early evolution, the shoreline will be displaced away from the Olkiluoto site (Fig. 7) due to continuing land uplift, even in case of global average sea level rise resulting from climate warming. As a result the surface environment will change and the fresh water infiltration will change the groundwater composition.

Following the period of early evolution, the repository and its geological environment will evolve to a quasi-steady target state, in which the engineered barrier performance targets and the host rock target properties are met and in which key safety-relevant physical and chemical characteristics are subject to relatively slow processes. By definition, the repository will be saturated and heat output of the fuel will have declined to a level that has no significant effect on the evolution of the repository system.

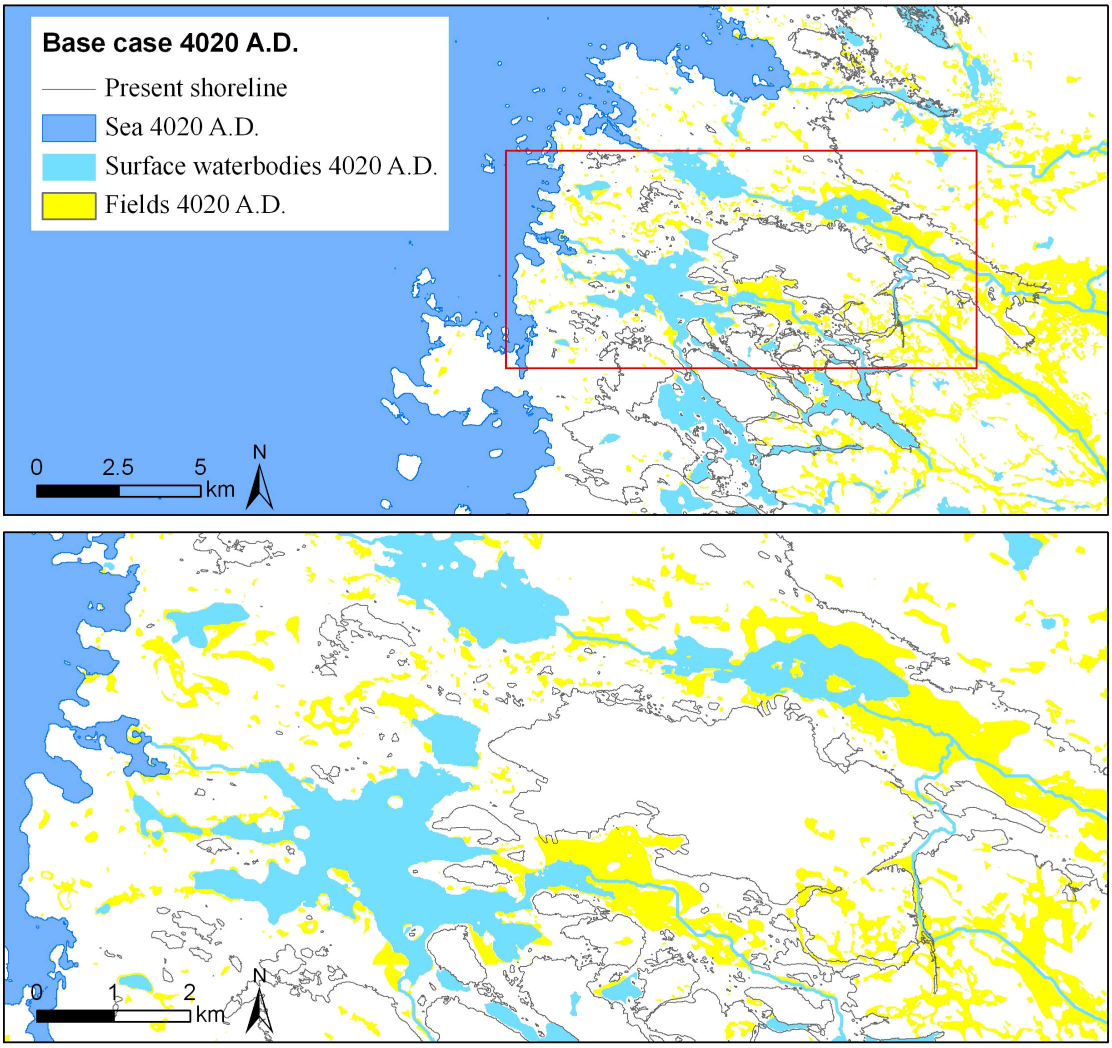

Figure 7. *Predicted evolution of the sea level, croplands and surface water bodies over the next two thousand years, and the present coastline as a gray line (Ikonen et al. 2010), map layout by Jani Helin, Posiva Oy.*

In the longer term, disposal system evolution could be affected by any future major climatic changes and, in particular, by the development of permafrost and ice sheets. Current understanding of climatic evolution indicates that, provided anthropogenic effects are negligible, the next few tens of thousands of years will be characterised by alternating permafrost and temperate climate phases, followed by a glacial period during which an ice sheet will cover the site. Anthropogenic emissions, especially greenhouse gases, could significantly delay the next permafrost stage and the future occurrence of ice sheets. Irrespective such uncertainties, permafrost and glaciation are not expected over the period for which dose calculations are required by regulations (the first several thousands of years). However, extended periods of permafrost and glacial episodes are expected to occur in the longer term and that these have to be taken into consideration in the safety case. Scientific understanding on climate development has increased considerably since these scenarios were formulated. Posiva has thus recently started a new project to reassess the time windows for warm and cold periods, their probabilities and possible extremes, in different climate scenarios covering the next 100 000 years, for use in future safety assessments

The formation of ice sheets could result in large water pressures and hydraulic gradients in the subsurface. The retreat and melting of ice sheets would result in the possibility that large volumes of meltwater could be forced into deeper parts of the bedrock. Furthermore, post-glacial earthquakes may occur following the retreat of the ice sheet, giving rise to stress changes in the rock that trigger shear movements on smaller-scale fractures that intersect the deposition drifts, tunnels and holes.

The repository is designed to tolerate such processes, and, in the current base scenario, the radionuclides associated with spent fuel are assumed to remain totally contained within the canisters over a prolonged time frame. Some of the potentially disruptive events associated with climate change, however, give rise to repository assessment scenarios involving canister failure and radionuclide release, as described below.

Formulation of repository assessment scenarios

Repository assessment scenarios explore the consequences of various uncertain features and perturbing processes that could potentially lead to canister failure, or significantly degrade the capacity of the repository system to limit radionuclide transport in the event of canister failure. Relevant features and processes are those that have the potential to compromise the capacity of the repository system to meet its performance targets, or to significantly perturb the target properties of the bedrock.

The following potentially relevant features and processes have been identified as arising ***internally within the disposal system***.

a. The possible presence of penetrating and non-penetrating defects in the canisters or other defects that could lead to early releases.

The presence of non-detected penetrating defects or other defects that could lead to early canister failure cannot currently be excluded. The probability of occurrence of one or more such defects is not yet quantified, but will be reduced by, for example, suitable weld inspection and quality control procedures. A performance target is that the canister has complete copper coverage over its entire surface for hundreds of thousands of years. Defective canister scenarios in which this target fails to be met are considered in the 2009 KBS-3V safety analysis and the KBS-3H safety assessment.

b. Processes leading to missing, loss or redistributed buffer mass.

Performance targets on buffer density set the range over which the buffer safety functions are expected to operate. Internal processes that could potentially compromise these targets include initial misemplacement of the buffer and erosion of the buffer by transient water flows ("piping") during saturation. The likelihood or extent of these occurrences will be reduced, for example, by appropriate quality control procedures and by applying appropriate rock suitability criteria for locating deposition holes. Furthermore, pessimistic scoping calculations indicate that loss of density may lead to corrosion failure of the canisters in less than 100 000 years only in the event of relatively high sulphide concentrations in groundwater. Nevertheless, a scenario in which the canister fails due to disruptive events affecting the buffer is considered in the 2009 KBS-3V safety analysis, and several calculation cases with perturbed radionuclide transport properties in the buffer surrounding a canister with an initial penetrating defect are considered both in the KBS-3H safety assessment and KBS-3V safety analysis.

c. Processes leading to perturbation of the buffer / rock interface.

Processes leading to perturbation of the buffer / rock interface include (i), the formation of an excavation damaged zone (EDZ), (ii), the occurrence of excavation- or thermally-induced rock spalling, and (iii), interaction of the buffer with cement leachates and with the Fe(II) from the corrosion of the KBS-3H supercontainer shells. Scoping calculations indicate that perturbations will have only minor effects on canister corrosion. However, increased mass transfer across the buffer / rock interface around a canister with an initial penetrating defect is considered in the KBS-3H safety assessment. Calculation cases in which the flow at the interface is varied are also analysed in the 2009 KBS-3V safety analysis.

d. Gas generated internally within the canister.

Significant volumes of gas will be generated by the corrosion of the cast iron canister insert following canister failure. Some of the C-14 released from the spent fuel can partition into a free gas phase and mix with corrosion-generated gas inside the canister, before being expelled once the gas pressure is sufficiently high. Depending on the canister corrosion rate, on the rate of supply of water from the buffer, and on the location of the point of failure in the copper overpack, another possibility is that water enters the canister interior, dissolves radionuclides and is then forced out of the canister once the gas pressure exceed the confining pressure of the bentonite buffer. These processes are addressed in a scenario analysed in the 2009 KBS-3V safety analysis, and in calculation cases of the KBS-3H safety assessment.

e. Criticality.

Based on the investigations carried out to date, and assuming credit for burnup to be taken, sustained induced fission (criticality) is not expected to occur in any of the types of canisters and for any of fuels that will be disposed of in a Finnish repository Scenarios including criticality have not been considered qualitatively in safety analyses. The possibility of criticality is not, however, completely ruled out, and will be considered further in future safety studies.

The following potentially relevant features and processes arise from events ***external to the disposal system***.

a. Buffer freezing.

Permafrost penetration to repository depth and consequent freezing of the buffer is considered unlikely. The possibility of buffer freezing is not, however, completely ruled out, and will be considered further in future safety studies. In safety analyses to date, although not considered explicitly as a scenario, the damage to the buffer safety functions that freezing could cause has been implicitly taken into account in calculation cases where the parameters of the buffer have been selected to address disturbances to it.

b. Canister failure due to isostatic load.

The highest isostatic loads on the canister will occur in association with future glaciations. Canister failure has, however, been ruled out on the basis of insert strength measurements, and this scenario has thus not been considered quantitatively in safety analyses.

c. Migration of oxygen to repository depth.

A target property of the bedrock is that conditions should be reducing, with no dissolved oxygen. Oxygen penetration to repository depth - for example in association with an influx of glacial meltwater - is judged unlikely, on the basis, for example, of hydrogeochemical and mineralogical site data that indicate that this has not occurred in the past. This process has not been considered as a potential cause of canister failure in safety analyses, although it has not been completely ruled out and will be evalutated further in ongoing studies. The 2009 KBS-3V safety analysis and the KBS-3H safety assessment have both considered the impact of glacial groundwater chemistry on radionuclide migration subsequent to canister failure by some other cause.

d. Loss of buffer due to exposure to glacial meltwater.

If the smectite clays in the buffer come into contact with water of low ionic strength, it is possible that the clays will be suspended as colloids and transported away from the deposition holes or drifts in flowing groundwater. A further target property of bedrock is that groundwater has sufficiently high ionic strength to avoid this “chemical erosion” of the buffer. A transient reduction in the ionic strength of groundwater at repository depth in association with glacial retreat and the penetration of dilute glacial meltwater into the bedrock is considered possible at Olkiluoto, and thus canister failure subsequent to buffer loss by chemical erosion has been considered as a scenario in the 2009 KBS-3V safety analysis and in the KBS-3H safety assessment.

e. Canister failure due to rock shear.

Finland in general and Olkiluoto in particular, has good tectonic stability, which is reflected by limited historical seismicity. The occurrence of earthquakes in the future cannot, however, be excluded, particularly in association with the retreat of ice sheets. Major features that are likely to be most affected by such earthquakes will be avoided by repository layout. Nevertheless, a large earthquake, even though unlikely at Olkiluoto, could trigger secondary shear movements on fractures intersecting deposition holes or drifts. These movements could lead to deformation of the bentonite buffer and to additional stresses being exerted on the canisters which, if sufficiently large, could lead to rupturing. The likelihood of such an event will be further reduced by applying rock suitability criteria (still under development). Nonetheless, canister failure by this mechanism is still considered possible and has been considered as a scenario in the 2009 KBS-3V safety analysis and in the KBS-3H safety assessment, as is also required by the Finnish regulatory guide.

Most of the above external processes are related to major climate change, although canister failure due to rock shear could potentially occur, though with very low probability, at any time.

Formulation of dose assessment scenarios

Dose assessment scenarios explore the consequences of the main uncertain features and processes that potentially could lead to alternative development and usage of the surface environment and migration paths for radionuclides into it. Potential radiological consequences to humans and other biota constitute the primarily endpoint to consider when identifying the main uncertain features and processes.

The dose assessment scenarios are driven primarily by climatic changes and the combination of climatic and land use changes. Thus, at Olkiluoto the shoreline position is time-dependent and may be inferred either from the land uplift rate alone, or from both the land uplift and sea level changes if the effect of climate change on sea level is taken into account. Land use is mainly driven by anthropogenic influences, here denoted *future human activities* (FHA); these can be included in the scenarios. FHA could to some extent affect conditions below the surface environment that can be taken into account in the variation of parameters used in the calculation cases derived from the repository assessment and or/dose assessment scenarios.

Formulation of human intrusion scenarios

Regarding human intrusion scenarios, Posiva holds the view that it is exclusively inadvertent - rather than deliberate - human intrusion that falls within the scope of the safety case. The Olkiluoto site has few resources that might attract deep drilling activities that could cause disturbance or damage, and human intrusion scenarios have not been analysed in the 2009 KBS-3V safety analysis and in the KBS-3H safety assessment. Human intrusion scenarios will, however, be formulated and analysed in future safety studies.

Analysis of scenarios in the 2009 KBS-3V safety analysis

The repository assessment scenarios analysed in the 2009 KBS-3V safety analysis are divided into two groups.

Defective canister scenarios:

- DCS-I: delayed penetrating defect – radionuclide release starting at 10 000 years after repository closure;
- DCS-II: early penetrating defect – groundwater in contact with spent fuel at repository closure.

Additional scenarios:

- ADI-1: earthquake / rock shear: canister fails as a consequence of the sudden displacement of a fracture intersecting the deposition hole;
- AD-II: canister fails as a consequence of disruptive events affecting the buffer, e.g. misemplacement of the buffer, intrusion of dilute glacial melt water, etc;
- AD-III: gas expels water with instant release fraction and/or radionuclides in volatile form (C-14) from the canister and deposition hole; no credit is taken for any retention of radionuclides by the buffer and backfill.

In this safety analysis, each individual scenario defines a general setting in which the system evolves, and is further characterised in terms of one or more calculation cases illustrating the impact of individual uncertainties, or uncertainties in combination. The majority of calculation cases were defined for the defective canister scenarios.

A base calculation case, Sh1, is defined for the initial penetrating defect scenario (DCS-II). Figure 8 shows the evolution of a quantity termed the "overall release ratio" (or simply "release ratio" or "ratio") and the contributions of the most important radionuclides in the base calculation case. The overall release ratio is defined as the sum over all calculated radionuclides of nuclide-specific release ratios. A nuclide-specific release ratio is defined as the ratio of the activity release rate of a given radionuclide to the corresponding regulatory geo-bio flux constraint. The overall release ratio is the parameter used to test compliance with regulations in the period after several thousand years in the future (see below).

A comparison of the results of the base calculation case with those of other calculation cases illustrates the effects of specific scenario, model or parameter uncertainties. For example, the size of an initial penetrating defect is uncertain. Additional calculation cases were analysed, showing that the effect of increasing the defect size from 1 mm in the base calculation case to 4 mm corresponds to an increase in the release ratio by a factor similar to the ratio of the defect areas (factor of 16).

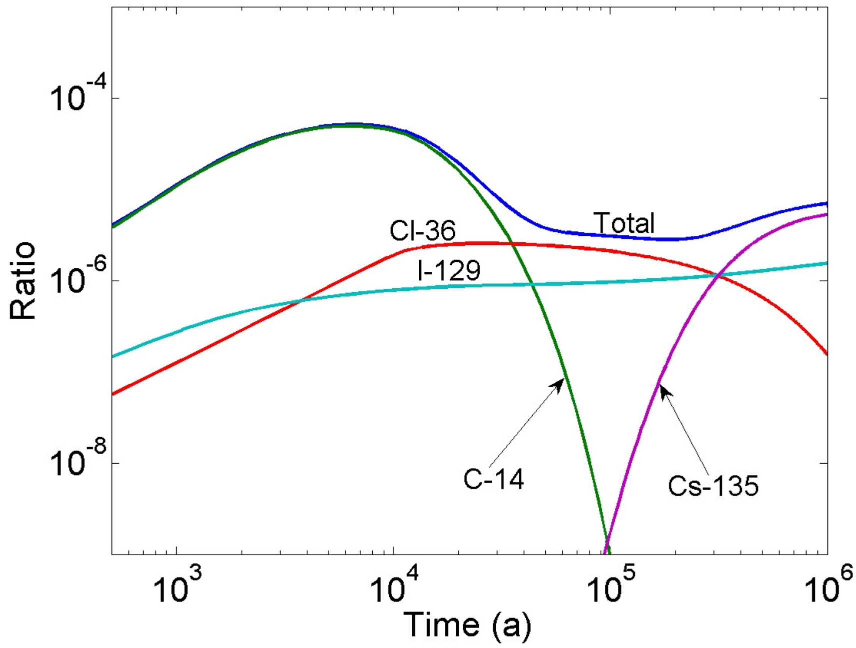

Figure 8*. KBS-3V safety analysis – base calculation case Sh1: Overall (total) release rate ratio as a function of time and the contributions of the most important radionuclides. The term "release rate ratio" is defined in the main text.*

Finnish regulations specify the quantitative dose and release criteria to be met by the repository. Dose assessments are explicitly required by regulations in cases where there are calculated releases to the biosphere within the time window from emplacement up to several thousand years in the future. Constraints are given on the average annual dose for the most exposed individuals and also for larger groups of people who may be exposed to radioactive releases. Thus, the annual dose maxima to representative persons for the most exposed group and to other exposed people, E_{group} and E_{pop}, have been calculated for a range of cases. The highest calculated value of E_{group} is 3×10^{-5} mSv, which is more than three orders of magnitude below the 0.1 mSv regulatory annual dose constraint for the most exposed individuals. The highest calculated value of E_{pop} is 5×10^{-6} mSv, which is more than two orders of magnitude below the 10^{-3} mSv lower limit of the regulatory annual dose constraint band for larger groups.

Finnish regulations state that disposal shall not affect other biota detrimentally, but no numerical constraints are given. The current Posiva methodology derives typical absorbed dose rates to assessment species (a group of species selected to cover different roles in the ecosystem) and compares the results with internationally proposed absorbed dose rate screening values for the protection of biota against radiation in the environment. Dose rates have been calculated for selected calculation cases; the maxima ranges from about 6×10^{-6} to 2×10^{-3} μGy/h for terrestrial species and from about 2×10^{-9} to 4×10^{-3} μGy/h for freshwater/marine species. These results show that the calculated dose rates are more than two orders of magnitude below the lowest proposed screening value. Thus, it is considered, with a high degree of confidence, that any releases from the repository do not affect species of flora and fauna detrimentally.

Constraints on the maximum radionuclide release rate across the geo-bio interface are also specified in Finnish regulations. According to the regulations, the sum of the ratios between the nuclide-specific activity releases and the respective constraints shall be less than one. The constraints are applicable at times beyond several thousand years in the future. Fig. 9 shows the maxima of the calculated overall release rate ratios and their times of occurrence for a subset of calculation cases considered in the 2009 KBS-3V safety analysis. The highest value of release rate ratio occurs in the calculation case RS1 of the rock shear/earthquake scenario. In this "what-if" case, the maximum release rate ratio for a single failed canister is more than an order of magnitude below the regulatory constraint (the possibility of multiple canister failures is a key issue for further study, see below).

Analysis of scenarios for a KBS-3H repository

Calculation cases for the analysis of KBS-3H repository assessment scenarios have been organised somewhat differently than those for KBS-3V. In particular, they are organised according to the canister failure mode that they address:

- PD: canister with an initial penetrating defect;
- CC: canister failure due to copper corrosion; and
- RS: canister failure due to rock shear.

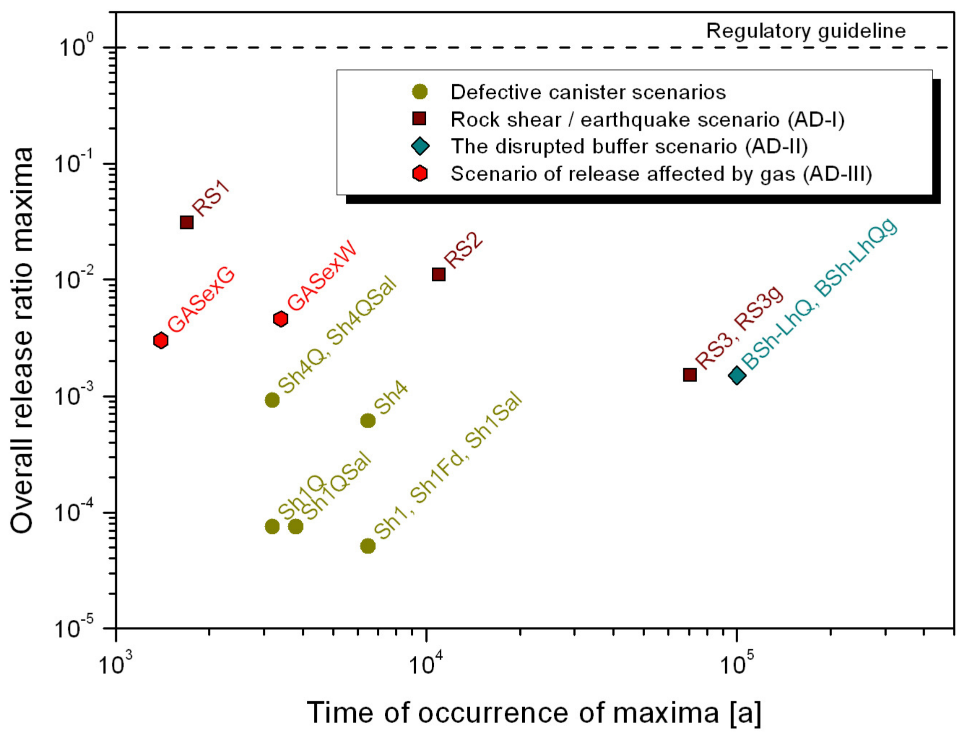

Figure 9. *The maxima of overall release rate ratio (as defined in the main text) and their times of occurrence in a selection of calculation cases for each scenario. Note that these cases represent a subset of those assessed in the 2009 KBS-3V safety analysis.*

For each canister failure mode, a base calculation case is defined, together with a number of variant calculation cases that illustrate the impact of individual uncertainties, or uncertainties in combination. As in the 2009 safety analysis of a KBS-3V repository, the majority of calculation cases were defined for a scenario in which a canister has an initial penetrating defect.

In general, the calculation results indicate that differences in the geometry and transport paths considered in the analysis of the KBS-3V and KBS-3H design variants have only a minor impact on calculated releases and doses. The focus of the analysis was on uncertainties that are specific to the KBS-3H variant, or have different implications for KBS-3H compared with KBS-3V. In this sense, the analysis was more limited in scope than in the 2009 safety analysis of a KBS-3V repository. The biosphere analysis for KBS-3H is also more limited than in the 2009 safety analysis of a KBS-3V repository, in the sense that only a single biosphere scenario has been applied, and compliance with all regulatory constraints was not fully assessed.

As noted above, there are a number of features and processes and that could significantly affect the characteristics of the buffer/rock interface. Some, such as thermally-induced rock spalling, are important for KBS-3V as well as KBS-3H, although the presence of the steel supercontainer shells and their corrosion products gives additional potential sources of perturbation for a KBS-3H repository. The impact of a perturbed buffer/rock interface on radionuclide releases to the biosphere in the event of a single canister failure is illustrated in Fig. 10 (case PD-SPALL addresses thermally-induced rock spalling PD-FEBENT1-3 address other potential processes, including iron-bentonite interactions).

A perturbed buffer/rock interface increases the calculated release ratio by no more than about an order of magnitude with respect to the base calculation case for the canister with an initial penetrating defect (PD-BC) in which an ideal, non-perturbed interface is assumed.

Another key issue for KBS-3H is expulsion of contaminated water from the interior of a failed canister by gas. Water entering the canister through the point of failure will corrode the canister insert. Any residual water will accumulate at the bottom of the internal void space within the canister. Corrosion will also generate gas, which will accumulate above this water. Gas pressure will increase until the gas eventually forces its way out of the canister through the point of failure. In so doing, it will also force out water, provided the water entering the canister is not entirely consumed by corrosion, and provided the point of canister failure lies below the gas-water interface. Water expelled in this manner may be contaminated with radionuclides. In the case that failure is due to the presence of a weld defect, this scenario is more plausible for KBS-3H than for KBS-3V. This is because the weld is near the top of the canister and, in KBS-3V, the canisters will be emplaced vertically upright. Any defect will therefore always be above the internal gas-water interface in KBS-3V, but not necessarily in KBS-3H.

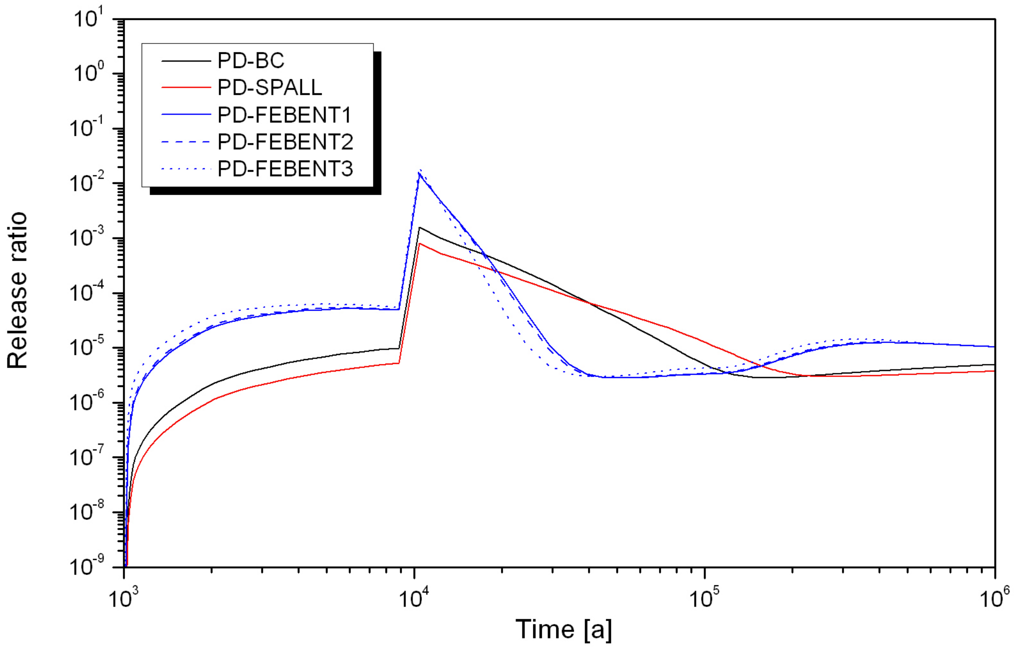

Figure 10. *Overall release ratios in all four calculation cases addressing perturbations to the buffer / rock interface and in the base calculation case (PD-BC) for the KBS-3H safety analysis.*

The calculated release rate maximum for this KBS-3H case (PD-EXPELL) is more than an order of magnitude higher than that of the base calculation case (PD-BC). This case also gave the highest annual dose to the most exposed *individual*, with a maximum of about 2×10^{-2} mSv occurring at about 5 000 years. This is a factor of 5 below the regulatory constraint of 0.10 mSv for the most exposed *persons*. For all other assessment cases, the annual dose maxima range from 5×10^{-4} to 3.5×10^{-3} mSv, and are thus around two orders of magnitude below the regulatory constraint, even though the annual doses as calculated are likely more conservative quantities than required for the comparison.

The highest calculated release ratio maximum occurred in a case addressing canister failure by copper corrosion, in association with an influx of glacial meltwater and loss of buffer mass by chemical erosion, coupled to an assumption of low transport resistance and saline geochemical conditions in the geosphere at later times. In this case, the release ratio maximum is more than an order of magnitude below the regulatory constraint. However, there are significant uncertainties associated with this scenario, e.g. whether substantial buffer mass loss by chemical erosion could occur at all, and, if it does, the number of canister positions that are likely to be affected. The development of a better understanding of chemical erosion is a priority for future work for both the KBS-3H and KBS-3V repositories.

Compliance with regulatory requirements

The regulatory requirements for the long-term safety of a geological repository in Finland are set out in detail in Guide YVL E.5, which is expected to be issued in 2010. Guide YVL E.5 is quoted throughout the present report, based on an unofficial English translation. It provides requirements and guidance related to conducting and documenting a safety analysis, compliance with which is described systematically in the main text.

YVL E.5 also specifies the quantitative dose and release criteria to be met by the repository. Compliance with these criteria has been described above for both KBS-3V and KBS-3H repositories, where mainly single canister failure cases have been considered. To date, only in the case of canister failure due to rock shear have the consequences of multiple canisters failures been estimated. The estimated geo-bio flux arising from multiple canister failures (estimated number of potentially affected canisters being in the order of 10 to 20 of 3000 canisters) in this scenario, which conservatively disregards the application of rock suitability criteria to avoid fractures with the potential to undergo damaging shear movements (criteria are still under development), nevertheless complies with the regulatory geo-bio flux constraint.

Evaluation and statement of confidence

The present report outlines Posiva's preliminary safety case for the final disposal of Finnish spent fuel in a geological repository at the Olkiluoto site. A full safety case for the KBS-3V variant of the KBS-3 method will be developed to support the Preliminary Safety Assessment Report (PSAR) in 2012. However, studies to date already indicate that, except a few unlikely circumstances affecting a small number of canisters, spent fuel is expected to remain isolated, and the radionuclides contained within the canisters, for hundreds of thousands of years or more, in accordance with the base scenario.

Confidence in the base scenario derives, in the first place, from the intrinsic properties of the main components of the repository and from the understanding of their evolution gained from extensive site- and concept-specific field, laboratory and modelling studies and from studies of natural and anthropogenic analogues. In particular, the canisters are mechanically strong and corrosion resistant. They are also protected by the surrounding bentonite buffer and by their deep underground location in rock that is geologically very stable and lacks resources that might attract deep drilling activities in the future that could disturb the repository.

Nevertheless, a small number of canister failures within a million year period cannot be excluded. The planned disposal system provides a series of barriers that delay and attenuate the releases from a failed canister, such that any exposure of humans and other biota to radioactivity is highly unlikely to cause harm. To demonstrate that this is the case, scenarios have been developed and analysed, taking into account the various uncertainties affecting the rates of release of radioactive substances from failed canisters and their subsequent migration through the geosphere and transport in the biosphere.

Regarding human intrusion, the geosphere lacks resources that might attract deep drilling activities in the future that could disturb the repository, but scenarios involving drilling into the repository cannot be excluded. The potential consequences of such scenarios involving drilling have not been evaluated in recent safety analyses, but will be evaluated as part of the 2012 safety case.

Quality measures have been applied in the development and application of models, data and computer codes in the assessments, including:

- 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;
- where possible, validation of the models used to analyse the scenarios; and
- verification of assessment codes.

Confidence in the findings of the assessments also derives from the systematic treatment of uncertainty in the safety analyses.

The low radionuclide calculated release rates to the biosphere and resultant annual doses imply that any radiological consequences of these releases will be negligible. Other safety indicators have also be used to place these results in perspective. The radiotoxicity flux, for example, is a measure of the hazard associated with a flux of radionuclides, and can be used to compare releases from the repository with radionuclide fluxes arising from other, natural sources (Fig. 11).

Beyond the million year time frame considered in the current base scenario, slow corrosion of the copper shell, the detrimental effects of multiple periods of glaciation or some other mechanism will eventually lead to failure of all the canisters and the release of some radioactivity to the surrounding rock. The radioactivity that is initially present in the repository will, however, decay to a much reduced levels before this happens.

Plans are in place to manage remaining safety-related issues and uncertainties with the aim of developing a final safety case in which the arguments for safety are insensitive to any residual uncertainties (see below).

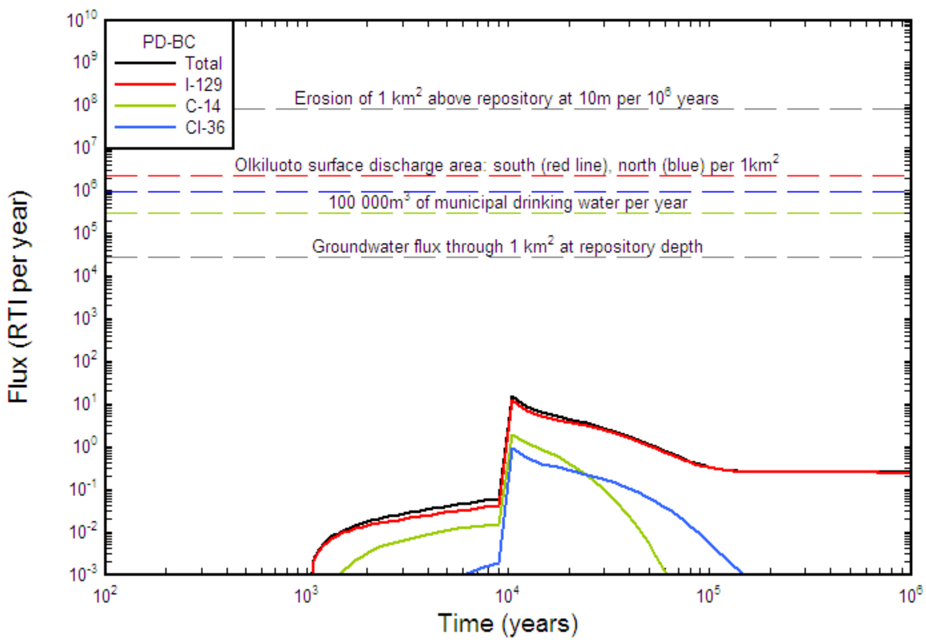

Figure 11. *Radiotoxicity flux from the KBS-3H repository into the biosphere for the penetrating defect base calculation case (case PD-BC in the KBS-3H safety assessment) compared with a range of naturally occurring radiotoxicity fluxes (see text and Appendix B, Neall et al. (2007) for further explanation).*

Based on these considerations, it can be stated that, in Posiva's view, there are good prospects that a geological repository for the final disposal of spent fuel, implemented as planned at the Olkiluoto site, will provide adequate levels of long-term safety.

The way forward

Posiva's iterative approach for the management of uncertainties can be summarised in four words: **identify**, **avoid**, **reduce** and **assess**. **Identification**, description, and where possible quantification of uncertainties, as well a consideration of their potential relevance to safety, represent an essential part of all the reports related to the development of the safety case. The development of the disposal system is based on the idea of robustness, which means, where practicable, **avoiding** concepts and components the behaviour of which is difficult to understand and predict, and **reducing** the impact of uncertainties – for example by introducing conservative safety margins in the design of some components. Some uncertainties will, however, always remain and have to be **assessed** in terms of their relevance to the final conclusions on safety.

The establishment of performance targets for the main system components provides guidance to robust system design, e.g. the range of saturated buffer densities that ensure that the performance targets for the buffer are achieved, and hence that the buffer safety functions are fulfilled. Safety assessment considers the impact of uncertainties on the assumption underlying the performance targets and on the capacity of the system to meet the targets as it evolves over time. In the safety analyses described in the present report, the deterministic approach to the identification and evaluation of calculation cases allows the impact of individual uncertainties on system performance and safety to be determined individually. This allows uncertainties that could potentially weaken the safety case to be identified, and avoided or reduced by research, technical design and development. A more systematic analysis of combinations of parameter uncertainties will be included in future safety assessments through a combination of probabilistic and deterministic approaches.

Current plans to address specific safety-related issues are given in the report TKS-2009, including issues regarding safety case development as well as more specific issues of scientific understanding. Key issues for safety case conceptualisation and methodological development include:

- how to handle the possibility of multiple canister failures;
- how to handle the combined effects of more than one disruptive event or process;
- how to ensure all relevant uncertainties are identified and their impacts assessed, including the potential use of probabilistic methods; and
- how to assure quality in the various steps in the production of the safety case.

The base scenario currently includes all lines of evolution of the disposal system giving no release of radionuclides. In future, those lines of evolution that lead to radionuclide release and that cannot be shown to have a low probability of occurrence may be included in the base scenario.

Many issues of scientific understanding are relevant to both KBS-3V and KBS-3H. While some issues, such as those related to gas generation prior to canister failure, are relevant mainly to KBS-3H, it should also be noted that there are some issues that are specific to KBS-3V, such as those related to the deposition tunnel, its backfill and its excavation damaged zone. The feasibility and quality of technical solutions will also be demonstrated by tests and experiments. In this way, a comprehensive safety case will be developed to support a final decision to implement the facility.

TABLE OF CONTENTS

PREFACE

This report was written by Paul Smith (SAM Switzerland GmbH) with contributions from the SAFCA group members Thomas Hjerpe (Saanio & Riekkola Oy) and Nuria Marcos (Saanio & Riekkola Oy). The progress of the study was supervised by SAFCA Core Group consisting of Ari Ikonen (Posiva Oy), Barbara Pastina (Saanio & Riekkola Oy), Margit Snellman (Saanio & Riekkola Oy), Marja Vuorio (Posiva Oy), Paul Smith (SAM Switzerland GmbH) and Pirjo Hellä (Pöyry Oyj).

Heini Laine (Saanio & Riekkola Oy) and Päivikki Mäntylä (Saanio & Riekkola Oy) helped in final editing of the report.

The report was reviewed in draft form by members of SAFCA Group and by the following individuals: Mike Thorne (Mike Thorne and Associates Limited, UK), Lawrence Johnson (Nagra, Switzerland), Paul Gierszewski (Nuclear Waste Management Organization, Canada), Mikko Nykyri (Safram Oy), Annika Hagros (Saanio & Riekkola Oy), Tapani Eurajoki (Fortum Power and Heat Oy) and by Markku Friberg (Posiva Oy).

1 INTRODUCTION

1.1 Background

1.1.1 Spent fuel production and management

According to the Nuclear Energy Act, promulgated in 1987 and including amendments up to 769/2004, nuclear waste - including spent nuclear fuel - generated in Finland must be processed, stored and disposed of in Finland. In 1995, the two Finnish nuclear power companies, Teollisuuden Voima Oyj (TVO) and Fortum Power and Heat Oy (Fortum), established Posiva Oy (Posiva) to implement the final disposal programme for spent nuclear fuel and to carry out the related research, technical design and development (RTD or TKS, in Finnish). Other nuclear wastes are handled and disposed of by the power companies themselves.

Finland currently has four commercial nuclear power units: two in Loviisa, owned by Fortum, and two in Olkiluoto (OL1 and OL2), owned by TVO. A further unit (OL3) is under construction at Olkiluoto. OL1 and OL2 are boiling water reactors (BWRs). OL3 is a European pressurised water reactor (EPR/PWR). The two Loviisa reactors (LO1 and LO2) are Russian-designed pressurised water reactors (VVER-440). TVO and Fortum plan to build further nuclear capacity at both Olkiluoto and Loviisa. Spent fuel will be stored at interim storage sites at Olkiluoto and Loviisa until a geological repository for final disposal of spent fuel is available and ready to begin operations.

In 2001, the Finnish parliament endorsed a Decision-in-Principle (DiP) whereby the spent fuel produced in the four current commercial nuclear power units at Loviisa and Olkiluoto will be disposed of in a geological repository at Olkiluoto, in the 1 800-1 900 million years old shield area of southern Finland. The location of the Olkiluoto site is shown in Figure 1-1.

A further DiP was made in 2002 that allows for the expansion of the disposal facility to accommodate fuel from OL3. These DiPs allow for the disposal of a maximum amount of spent fuel corresponding to 6 500 tonnes of uranium initially loaded into the reactor (tU). During 2008, Posiva submitted an application for a DiP to the government that allows an expansion to accommodate a maximum of 9 000 tU, which takes into account the planned fourth reactor at Olkiluoto. Subsequently, during 2009, Posiva submitted an application for a DiP that allows a further expansion to accommodate a maximum of 12 000 tU, which takes into account the planned third reactor at Loviisa. Safety studies summarised in the present report have been based on estimates of the amount of spent nuclear that will be produced by the four current commercial power units plus OL3: 5 550 tU in 2 820 canisters (Table 4-1 in Posiva 2009a). However, Posiva's Environmental Impact Assessment (EIA) of 2008 found that an expansion to a maximum capacity of 12 000 tU would have no significant additional long-term safety implications (Posiva 2008a).

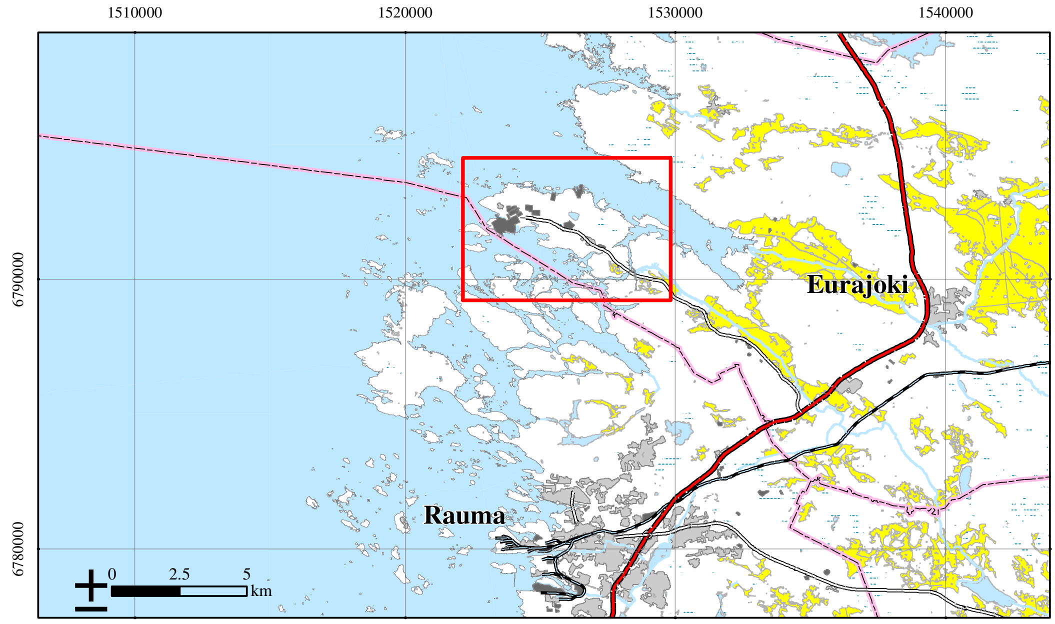

Figure 1-1. *The location of the Olkiluoto site. Base maps: ©National Land Survey, permission 41/MML/10.*

According to the decision of the Ministry of Trade and Industry (KTM) of 23 October 2003 (decision 9/815/2003), Posiva is to submit an application for a construction licence for a disposal facility at Olkiluoto by the end of 2012. In 2009, Posiva submitted the first outline version of the Preliminary Safety Analysis Report (PSAR) in support of construction license application. The outline PSAR will then be gradually updated to develop a full version that will support the actual licensing application. A Final Safety Analysis Report (FSAR) will be submitted at the time of the operational license application, in 2018. The target is to begin disposal operations in 2020.

1.1.2 The KBS-3 method

The 2001 DiP states that final disposal of spent fuel shall take place in a geological repository at the Olkiluoto site, developed according to the KBS-3 method. In the KBS-3 method, spent fuel encapsulated in water-tight and gas-tight sealed metallic canisters is emplaced deep underground in a geological repository constructed in the bedrock. In Posiva's current repository designs, the repository is constructed at a single level at a depth of about 400 m in the Olkiluoto bedrock.

Currently, two variants of the KBS-3 method are under consideration in both Finland and Sweden - KBS-3V and KBS-3H - as illustrated in Figure 1-2. As currently foreseen, a provision allowing for the implementation of either of the variants will be included in Posiva's planned 2012 construction license application, but the reference design will be based on KBS-3V. For the KBS-3H variant, more research, technical design, development and demonstration is needed to reach the same level of preparedness as for KBS-3V. According to current planning, the decision of which variant finally to implement can be made in the run up to the construction of the underground facilities in 2015-2016 (Section 2.4 of Posiva 2009a).

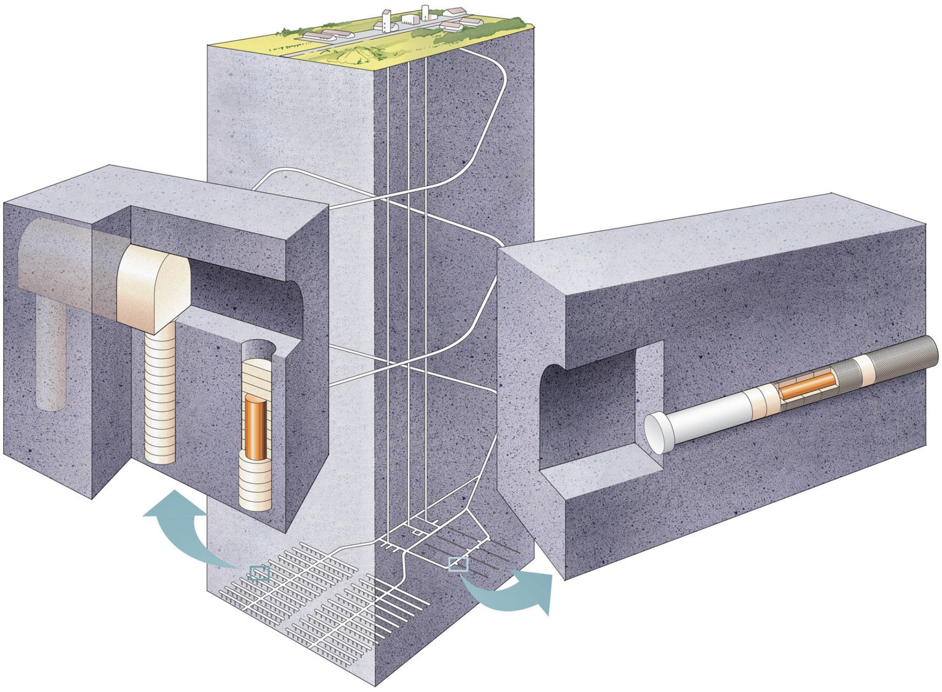

Figure 1-2. *The KBS-3V (left) and KBS-3H (right) alternative realisations of the KBS-3 spent fuel disposal method.*

In KBS-3V, the canisters are emplaced vertically in individual deposition holes constructed in the floors of deposition tunnels. In KBS-3H, several canisters are emplaced horizontally in a system of 100-300 m long deposition drifts. The KBS-3V deposition holes have a similar diameter to the KBS-3H deposition drifts (around 1.85 m). In both variants, the canisters are surrounded by a swelling clay buffer material that separates them from the bedrock and, in the case of KBS-3H, also separates the canisters one from another along the deposition drifts. The KBS-3V deposition tunnels and other underground openings in both variants are to be backfilled with a low permeability material.

1.1.3 Posiva's safety studies

The KBS-3 method was originally put forward by the Swedish waste management organisation (SKB). Posiva embraced the method more than 30 years ago and, since then, has been carrying out scientific research, technical design and development (RTD) to further refine the method, in close cooperation with SKB. Posiva's research, technical design and development have included around 20 years of site-specific investigations at the Olkiluoto site.

The long-term safety of a KBS-3 repository at Olkiluoto and other potential sites in Finland was assessed in the TILA-99 safety analysis report (Vieno & Nordman 1999) prepared by VTT Energy. The safety analysis was based on the same principles as the earlier TVO-92 safety analysis published in 1992 (Vieno et al. 1992) and the TILA-96 safety analysis published in 1996 (Vieno & Nordman 1996).

After the DiP in 2001, site studies have focussed exclusively on Olkiluoto. In June 2004, Posiva started building the underground rock characterisation facility - ONKALO - to enable it to undertake site-specific underground investigations. ONKALO may also be used as part of the future repository. On the basis of these site investigations and other research, technical design and development, Posiva will plan the repository in detail, prepare construction engineering solutions and assess safety.

Ongoing safety studies for a KBS-3V repository at Olkiluoto have resulted in production of a number of reports, including, most recently, the 2009 safety analysis of a KBS-3V repository. This includes the radionuclide transport study RNT-2008 (Nykyri et al. 2008), which assesses the potential magnitude and timing of any release of radionuclides from the repository to the surface environment, and BSA-2009 (Hjerpe et al. 2010), which includes modelling of the transport of these radionuclides within the surface environment and an assessment of the possible radiological consequences to humans and other biota. Posiva and SKB have also conducted a joint Research, Demonstration and Development (RD&D) programme in 2002-2007 with the overall aim of establishing whether KBS-3H represents a feasible alternative to KBS-3V. A safety assessment[2] for a KBS-3H repository at Olkiluoto has been published in 2008 (Smith et al. 2007a-c, Gribi et al. 2007, Neall et al. 2007 and Broed et al. 2007).

1.1.4 Posiva's safety case – its development and documentation

A safety case will be an essential element of both the PSAR supporting the construction license application and the FSAR supporting the operation license application. It has been defined by international organisations as a synthesis of evidence, analyses and arguments that quantify and substantiate the safety, and the level of expert confidence in the safety, of a geological disposal facility for radioactive waste (IAEA 2006, NEA 2004). A safety case includes a quantitative safety assessment, which is the process of systematically analysing the ability of the disposal facility to provide its safety functions and to meet long-term safety requirements, together with an evaluation of the potential radiological hazards and compliance with the safety requirements. The safety case, however, broadens the scope of the safety assessment to include the compilation of a wide range of evidence and arguments that complement and support the reliability of the results of the quantitative analyses.

Posiva's safety case for a spent fuel repository at the Olkiluoto site will be developed according to the plan published in 2008 (Posiva 2008b), which updates an earlier plan published in 2005 (Vieno & Ikonen 2005). In the new safety case plan, emphasis is put on quality assurance and control procedures and their documentation as well as consistent handling of different types of uncertainties. The quality assurance is based on a process approach that is consistent with the ISO 9001:2000 standard, but also takes into account specific regulatory aspects.

In developing the safety case, Posiva's iterative approach for the management of safety-relevant uncertainties can be summarised in four words: **identify**, **avoid**, **reduce** and

[2] The term safety analysis is used for the 2009 KBS-3V study, since this study focussed on the analysis of scenarios. The term safety assessment is used for the 2007 KBS-3H study, since this study was broader, including, for example, a Complementary Evaluations of Safety Report (Neall et al. 2007).

assess. **Identification**, description, and where possible quantification of uncertainties, as well a consideration of their potential relevance to safety, represent an essential part of all the reports related to the development of the safety case. The development of the disposal system is based on the idea of robustness, which means, where practicable, **avoiding** concepts and components the behaviour of which is difficult to understand and predict, and **reducing** the impact of uncertainties – for example by introducing conservative safety margins in the design of some components. Some uncertainties will, however, always remain and have to be **assessed** in terms of their relevance to the final conclusions on safety.

Posiva's safety case will be documented in a report portfolio, the main reports of which are shown in Figure 1-3 and described in more detail in Posiva (2008b). The *Description of the Disposal System Report* will summarise the information on the waste form, the engineered barrier system and the Olkiluoto site. More detailed descriptions will be given in technical and scientific reports on various components of the disposal system. The features, events and processes affecting the evolution of the repository will be described in the *Process Report* supported by a background report documenting lists or a database of features, events and processes (FEPs). The evolution of the repository and the scenarios for analysis in the safety assessment will be described in the *Formulation of Scenarios Report*. The *Models and Data Report* will document data and their interpretation (including modelling) in the context of the safety case. The assessment of the releases of radionuclides and the radiological consequences of these releases will be presented in the Analysis of Scenarios Report. A *Complementary Considerations Report* will provide additional evidence, arguments and analyses contributing to the safety case. Finally, the whole safety case, including the main results, will be described in a *Summary Report*. This report will provide input to the Preliminary Safety Analysis Report (PSAR) needed for the application for a repository construction license. It will also be updated to provide input to the Final Safety Analysis Report (FSAR) needed for the operational license application. The structures of the PSAR and the FSAR will be specified by STUK based on existing legislation and regulations. The Summary Report is, however, foreseen as forming a key element of both reports.

1.2 Regulatory context

The regulatory requirements for the long-term safety of a geological repository in Finland are set out in the Government Decree on the safety of disposal of nuclear waste (DG 736/2008) and, in more detail, in Guide YVL E.5, which is expected to be issued in 2010 by STUK, the Finnish Radiation and Nuclear Safety Authority, and will supersede the earlier YVL 8.4 issued in 2001 (STUK 2001). Guide YVL E.5 will cover all aspects of the disposal of nuclear waste, including spent nuclear fuel. It will cover radiation protection during the operation of the disposal facility and long-term safety. In an appendix, it will also provide guidance on regulatory expectations from the safety case. Guide YVL E.5 is quoted throughout the present document, based on an unofficial English translation of draft version 3 (STUK 2009).

1.3 Aims, scope and structure of this report

The present report is an interim version of the Summary Report of the safety case portfolio. The production of an interim version is seen as useful in that it provides a platform for discussion regarding:

- the structure of the safety case;
- the types of evidence, arguments and analyses to be presented;
- the planned methodology for the safety case to be presented in the PSAR; and
- the present level of maturity of the safety case with respect to that needed for the construction licence application.

The current version of the Summary Report is based on the results of safety studies carried out to date, and is acknowledged to have some important limitations. These include the lack of an analysis of human intrusion scenarios, which are required to be considered according to Finnish regulations, and the lack of an analysis of the rate or probability of canister failure by different modes. The analyses documented in this interim version of the Summary Report, and especially the limitations and knowledge gaps identified in conducting these analyses, have provided key input to the three-year research, technical design and development plan for nuclear waste management of the Olkiluoto and Loviisa nuclear power plants, TKS-2009 (Posiva 2009a). The findings of

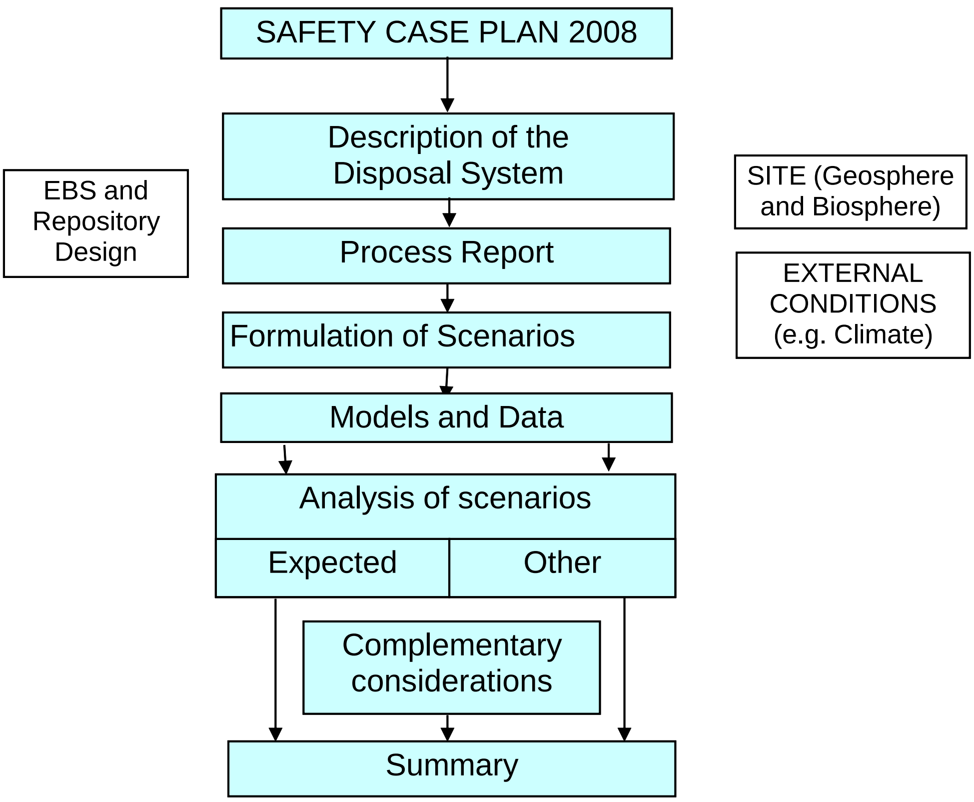

Figure 1-3. *Main reports of the safety case portfolio (in blue) and the main input from supporting technical and scientific activities (in white). EBS: engineered barrier system.*

this programme will in turn be incorporated in the Summary Report as it updated for the PSAR and the FSAR.

The scope of this report addresses the potential long-term radiological hazard to human health and to the environment due to the repository. Issues outside the scope of this report include operational safety and the potential hazard due to chemically toxic material present in the repository.

The structure of the report is illustrated in a “roadmap” figure that appears at the start of each of the following chapters (Fig. 2-1 and similar figures in the chapters thereafter).

- The present chapter (Chapter 1) introduces the context and objectives of the report as a whole.
- Chapter 2 describes the hazards presented by spent fuel, the need for safe disposal of spent fuel and the safety concept for disposal, which is a description of how safe disposal of spent fuel is achieved with the KBS-3 method, and the characteristics of the Olkiluoto site.
- Chapter 3 presents a description of the main components of the disposal system for which a safety case is to be made, including both the KBS-3V and KBS-3H variants, and of how the repository will be implemented.
- Chapter 4 presents a description of the assessment basis, which is the scientific and technological information, understanding, models and data on which the safety assessment and safety case are based, and the methodology for carrying out the safety assessment (it should be noted that the description of the main components of the disposal system given in Chapter 3 may also be considered a part of the assessment basis).
- Chapter 5 presents a description of the base scenario for the evolution of a KBS-3V or KBS-3H repository at the Olkiluoto site.
- Chapter 6 deals with the formulation of assessment scenarios, including repository assessment scenarios and dose assessment scenarios (human intrusion scenarios, which are also classified as assessment scenarios, will be formulated and analysed in future safety studies).
- Chapter 7 presents a summary of the results of the 2009 safety analysis of a KBS-3V repository.
- Chapter 8 presents a summary of the results of the analysis of scenarios in the KBS-3H safety assessment.
- Chapter 9 discusses compliance with Finnish regulatory guidance on the long-term safety of geological disposal of spent fuel, as set out in the regulatory Guide YVL E.5.
- Chapter 10 summarises the main evidence, arguments and analyses that lead to confidence in the safety of a KBS-3V or KBS-3H repository at the Olkiluoto site.
- Finally, Chapter 11 describes the broad strategy for identifying and managing safety-related issues that will have to be addressed prior to the compilation of the PSAR and the FSAR, and lists the main issues.

Some key modelling assumptions for safety analysis and their classification are presented in an appendix to this report.

2 THE SAFETY CONCEPT

This chapter (Fig. 2-1) describes the hazards presented by spent fuel, the need for safe disposal of spent fuel and the safety concept for disposal, which is an overall description of how safe disposal of spent fuel is achieved.

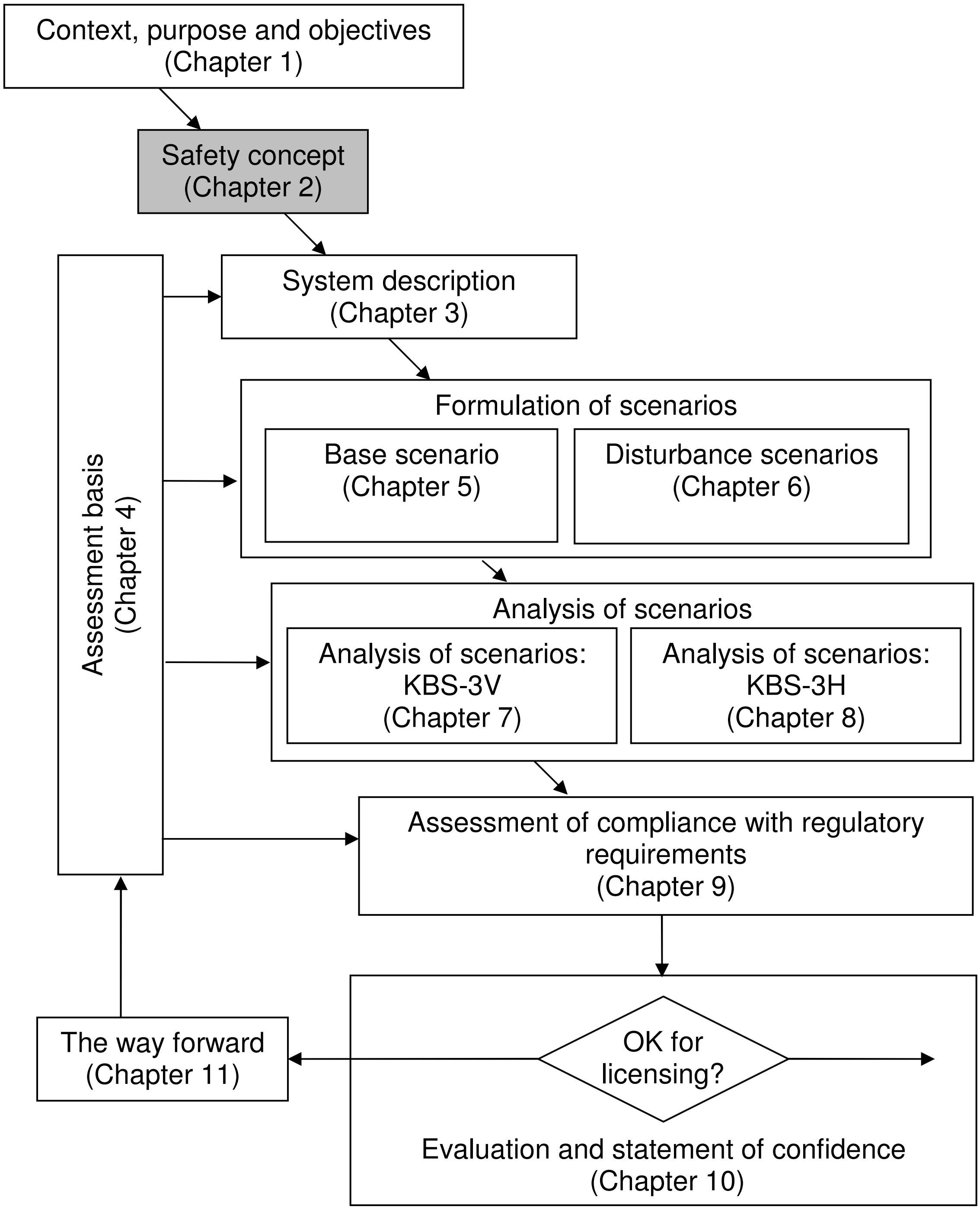

Figure 2-1. *The present chapter in the context of the safety case summary report.*

The chapter is structured as follows:

- Section 2.1 describes the nature and evolution of the hazards presented by spent fuel and the need for isolation and containment;
- Section 2.2 presents the main elements of the safety concept that provide this long-term isolation and containment in the KBS-3 method; and
- Section 2.3 discusses the implementation of the safety concept in the KBS-3V and KBS-3H variants.

2.1 Nature and evolution of the hazards presented by spent fuel

Both humans and the environment have to be protected from the hazards presented by spent fuel. These hazards have been discussed in numerous documents published by the radioactive waste management community. Particularly useful discussions are provided in NEA (2007), Hedin (1997) and Neall et al. (2007).

The radiological hazard that spent fuel presents to human health and to the environment decreases significantly over time due to radioactive decay. Various illustrations of the reduction of radioactivity and associated radiological hazard over time are given in Neall et al. (2007). It can be shown, for example, that each tonne of spent fuel becomes comparable, in terms of its radiological toxicity on ingestion, to the 8 tonnes of natural uranium from which the precursor fuel was derived on a timescale in the order of a hundred thousand years (see e.g. Fig. 2-4 of Neall et al. 2007). Figure 2-2, which is taken from Hedin (1997), shows that external gamma and neutron radiation dose rates received by a person in close proximity to spent fuel also decline significantly over time, although even after a hundred thousand years, a significant gamma dose rate would still be received. In NEA (2007) it is remarked:

"*… even though the hazard potential of spent fuel and some long-lived wastes decreases markedly over time, these wastes can never be said to be intrinsically harmless*."

However, as long as the spent fuel is isolated deep underground in a closed geological repository and the radionuclides associated with the spent fuel are contained, the shielding that this system provides is such that there is no radiological hazard to humans.

2.2 Main elements of the safety concept

The KBS-3 method provides long-term safety by isolating spent fuel deep underground, and containing its radionuclides within a system of multiple barriers, both engineered and natural, which ensure that no single harmful event or deficiency in one of the barriers may endanger the ability of the system to provide safety. The safety concept whereby long-term isolation and containment are achieved is illustrated in Figure 2-3. Containment of the radionuclides is provided first and foremost by encapsulating the spent fuel in sealed canisters. The figure shows as red pillars and blocks key features of the system supporting the operation of this safety function. The figure also shows as green pillars and blocks secondary features of the system that ensure the retention of

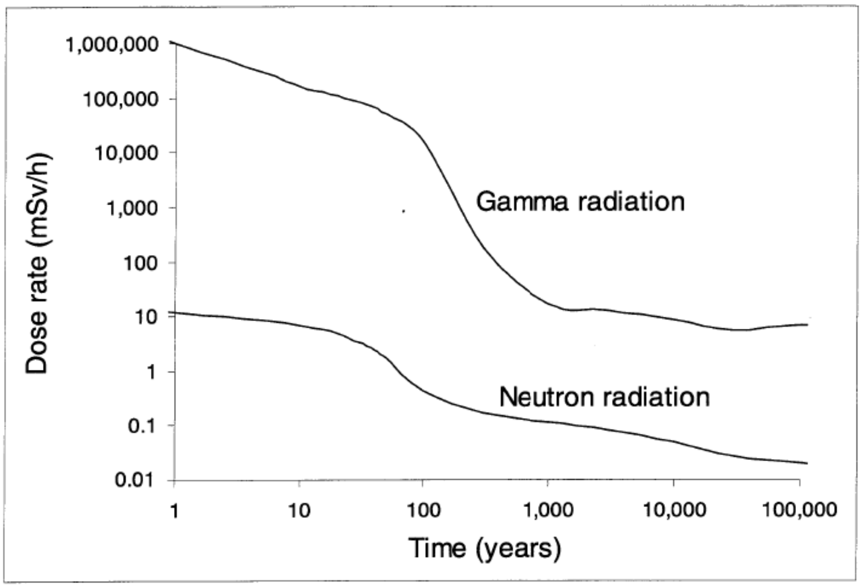

Figure 2-2. *Dose rate at a distance of 1 metre from one tonne of spent nuclear fuel as a function of time after discharge from the reactor (after Fig. 3-8b in Hedin 1997).*

radionuclides and the limitation and retardation of radionuclide releases in the event of canister failure. These primary and secondary features of the safety concept are described in the following text (and indicated in bold italics).

As noted in Section 1.1.4, safe disposal relies on the adoption of a ***robust system design***, A robust system design is derived through an iterative process in which features, events and processes (FEPs) potentially affecting the safety functions performed by the repository barriers are first identified and then the related uncertainties are avoided, reduced, or their effects mitigated by research, technical design and development (see the discussion in Chapter 11). The resulting robust design is one in which all reasonably foreseeable perturbing phenomena and remaining uncertainties can be shown not to compromise long-term safety.

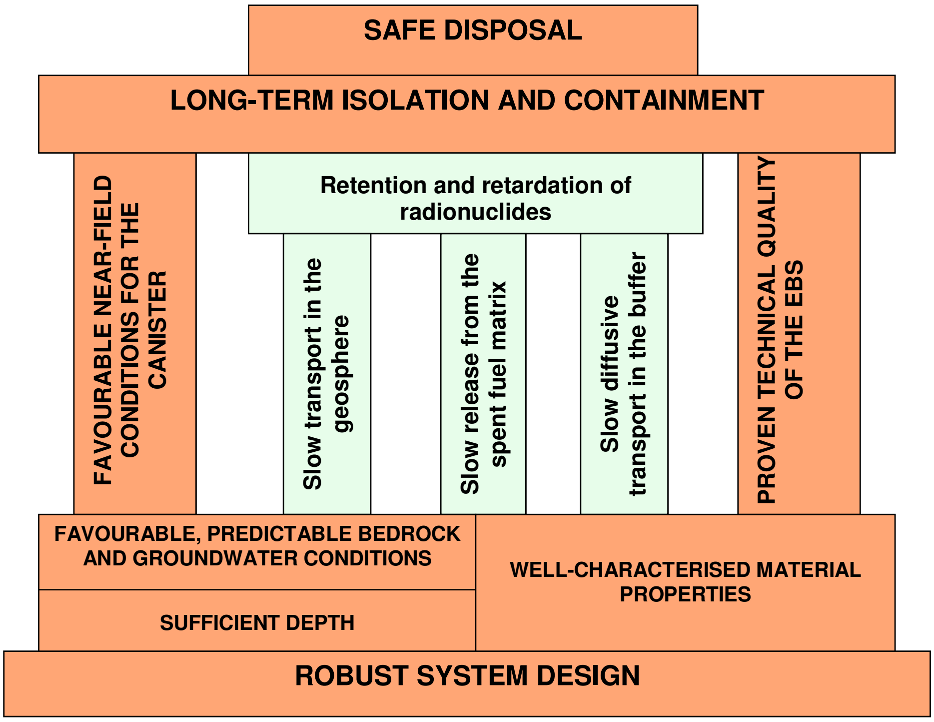

Figure 2-3. *Outline of the safety concept for a KBS-3 type repository for spent fuel in a crystalline bedrock.*

For a design to be robust, no single detrimental phenomenon or uncertainty should have the potential to undermine all the safety functions provided by the system components. This is achieved in the KBS-3 method through the provision of multiple barriers and a range of at least partly independent features and processes that contribute to their safety functions. Thus, for example, features and processes associated with the green columns in Figure 2-3 are at least partly independent of those shown in red. The safety functions assigned to various system components in the KBS-3 method are described in Section 2.3.

Other elements of a robust system design include ***sufficient depth*** for the repository, ***favourable and predictable bedrock and groundwater conditions*** and ***well characterised material properties*** of both the bedrock and the engineered barrier system (EBS). The characterisation of the Olkiluoto site and the strategy for repository design are focused on a volume of bedrock situated between 400 and 500 metres below the ground surface. At such depths, favourable and predictable bedrock and groundwater conditions are found and the likelihood of inadvertent human intrusion is low. The depth range is consistent with guidance in YVL E.5, according to which the repository should be:

"*at a sufficient depth in the bedrock in order to mitigate the impacts of above-ground events, actions and environmental changes on the long-term safety and to render inadvertent human intrusion to the repository very difficult*" (YVL E.5, Para. 3.3).

The likelihood of canister failure and radionuclide release is kept small by providing an environment around the canisters that favours their longevity (***favourable near-field conditions for the canisters***) and by the ***proven technical quality of the EBS***. The EBS includes the canister itself, a surrounding clay buffer that protects the canisters from small rock movements and from potential detrimental substances in the groundwater, and a low-permeable material used as backfill in KBS-3V deposition tunnels that further protects the deposition hole from detrimental processes.

Finally, should any initially defective canisters be present or subsequent breaches in the canisters occur, the consequences of radionuclide releases to humans and other biota inhabiting the surface environment are mitigated by the ***slow release from the spent fuel matrix, slow diffusive transport in the buffer, and slow radionuclide transport in the geosphere***. Together, these provide **retention and retardation of radionuclides**, such that radioactive decay reduces any potential release to the surface environment. These features of the safety concept are depicted in Figure 2-3 as secondary aspects of the safety concept (green blocks and pillars) since they become important only in the event of canister failure.

2.3 Implementation of the safety concept

2.3.1 Safety-related design requirements

All the components of the repository must satisfy requirements that are defined in Posiva´s requirements management system (VAHA). The *VAHA programme and database* has been established to define, document and manage the various types of requirement placed on each component of the repository stemming from different origins (e.g. scientific, technical, socio-economic). Safety-related design requirements aim to ensure, as far as possible:

- mutual compatibility of the engineered barriers with each other and with the bedrock, taking into account their respective safety functions;
- resistance of the engineered barriers to the main thermal, hydraulic, mechanical and chemical loads to which they will be subjected as the system evolves; and
- robustness with respect to slow changes and infrequent events that may occur over the course of time.

The current requirements for the KBS-3V barriers and those barriers that are common to both the KBS-3V and KBS-3H variants are described in detail in the research, technical design and development programme for 2010-2012: TKS-2009 (Posiva 2009a). The requirements for the KBS-3H barriers will be presented in detail in the reporting of the ongoing Complementary Study Phase in 2010. These are being further developed iteratively, along with repository design and safety assessment.

2.3.2 The KBS-3V and KBS-3H design variants

In current designs for both KBS-3V and KBS-3H, each canister includes a cast iron insert enclosed entirely within a copper overpack. In the KBS-3V design variant, individual canisters with a surrounding compacted bentonite clay buffer are emplaced in deposition holes constructed in the floors of deposition tunnels that are subsequently backfilled (Fig. 2-4). The buffer is chemically and mechanically stable and swells as it takes up water. Thus, the initial gaps between the canister and the buffer and between the buffer and rock become filled with bentonite over time as water migrates into the buffer from the surrounding rock and overlying backfilled tunnel.

In the KBS-3H variant, each canister is pre-packaged in an assembly, called a “supercontainer”, before emplacement in the deposition drifts. In the current design, the supercontainer consists of a perforated steel[3] shell cylinder containing the canister surrounded by a layer of bentonite (Fig. 2-5). The purpose of the supercontainer is to facilitate emplacement operations. Bentonite distance blocks separate adjacent supercontainers one from another along each drift. Figure 2-6 shows a section of a KBS-3H drift with two supercontainers and one distance block.

The KBS-3H buffer comprises the bentonite inside the supercontainers and the bentonite of the distance blocks. Over time, the buffer swells to fill gaps between the supercontainers and the distance blocks, and between each of these and the drift wall. Supercontainers and distance blocks are not emplaced in deposition drift sections where groundwater inflow might otherwise lead to detrimental effects on the buffer (in

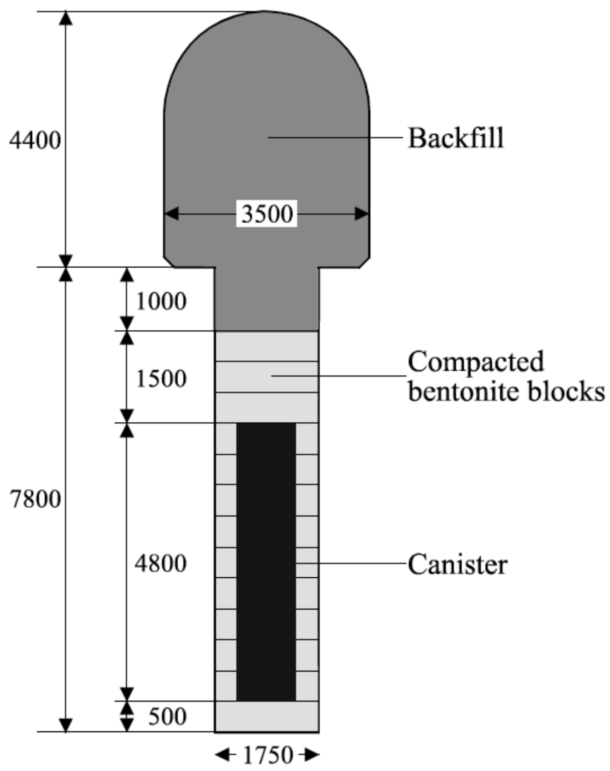

Figure 2-4. *Schematic view of a KBS-3V deposition hole. Dimensions are for a BWR fuel canister (OL1 and OL2 fuel) (Fig. 3-2 in Vieno & Nordman 1999).*

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[3] Alternative materials for the shell, such as titanium and copper, are being studied, since these may reduce the potential for detrimental interactions with the buffer.

particular erosion of the buffer by transient water flows, including “piping”, during saturation[4]). Instead, depending on the rate of inflow, either filling blocks are inserted in these sections, or they are hydraulically isolated from the rest of the drift by means of compartment plugs. Filling blocks and compartment plugs are described further in Autio et al. (2008).

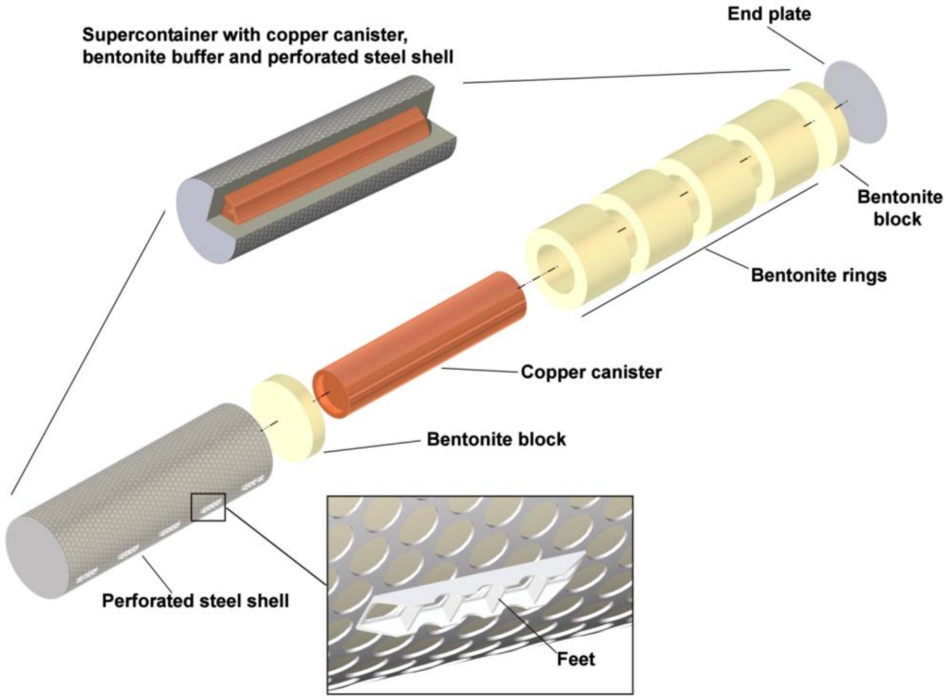

Figure 2-5. *Exploded view of the KBS-3H supercontainer (Fig. 3-3 in Autio et al. 2007).*

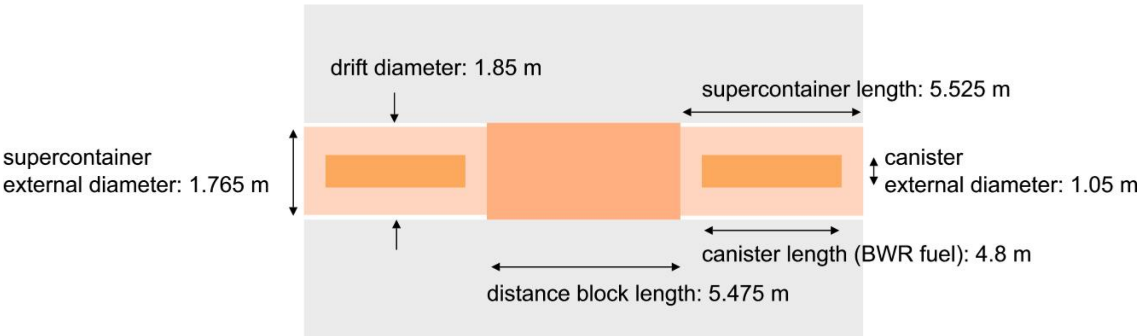

Figure 2-6. *Illustration of a section of a KBS-3H deposition drift with two supercontainers separated by a distance block. The 5.475 m distance block length is for BWR fuel (OL-1 and OL-2 fuel) (Fig. 1.4-3 in Smith et al. 2007c).*

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4 Börgesson & Sandén (2006) and SKB (2006a) have summarised the present knowledge of erosion due to transient water flows, including “piping”, during early evolution (see also Process 5.2.3 in Miller & Marcos 2007). Piping occurs when the transient water flows take place in discrete channels within the buffer. As described in TKS-2009, erosion due to piping is one of the most significant issues identified in Posiva’s BENTO programme.

Again because of concerns regarding the possibility of piping and erosion, and also the possibility of mechanical displacement of the distance blocks caused by hydraulic pressure differences during repository saturation, versions of KBS-3H termed DAWE (Drainage Artificial Watering and air Evacuation) and STC (Semi Tight Compartments) have been developed in which open spaces in the KBS-3H deposition drifts are filled with artificially injected water, rather than relying on natural groundwater inflow (Autio et al. 2008).

2.3.3 Repository components and their safety functions

The repository consists of a set of engineered and natural barriers. The roles that these barriers play in achieving safe disposal are called their safety functions. The main barriers in the KBS-3V variant, which all serve important safety functions, are:

- the canister;
- the buffer;
- the backfill in the deposition tunnel; and
- the host rock.

Barriers specific to the KBS-3H alternative and their safety functions are being considered during the ongoing Complementary Study Phase. Below, only those KBS-3H barriers that are also KBS-3V barriers, namely the canister, the buffer and the host rock, are discussed.

A prolonged period of complete containment of radionuclides is the main safety function of the canister. This safety function rests first and foremost on the mechanical strength of the cast iron insert and the corrosion resistance of the copper surrounding it.

The safety functions of the buffer are (a), to contribute to mechanical, geochemical and hydrogeological conditions that are predictable and favourable to the canister, and to protect the canisters from external processes that could compromise the safety function of complete containment of the spent fuel and associated radionuclides and (b), to limit and retard radionuclide releases in the event of canister failure. In case of the KBS-3H variant, the buffer has the additional safety function of separating the supercontainers hydraulically one from another, thus limiting the possibility of preferential pathways for flow and advective transport along the buffer/rock interface. This is required because the buffer/rock interface near to the canisters may locally be perturbed by a number of processes, as described in later chapters of this report (see, e.g. Section 8.2.2).

The safety functions of the KBS-3V deposition tunnel backfill are (a), to contribute to favourable and predictable mechanical, geochemical and hydrogeological conditions for the buffer and canisters and (b), to limit and retard radionuclide releases in the event of canister failure. It also (c), contributes to the mechanical stability of the rock adjacent to the deposition tunnels.

The safety functions of the host rock are to provide (a), isolation of the spent fuel from the surface environment and normal human habitat and limit the possibility of human intrusion, (b), favourable and predictable mechanical, geochemical and hydrogeological conditions for the engineered barriers, and (c), a barrier that limits and retards the migration to the biosphere of radionuclides released from the repository.

Implementation of the KBS-3V and KBS-3H variants also entails the introduction of a number of auxiliary components. These components are required for reasons of engineering practicality, but in some cases also have safety functions. Safety functions arise, for example, because implementation involves the construction of a system of underground openings, including access ramps and shafts, that would significantly perturb safety functions of the host rock unless backfilled and sealed. Auxiliary components of the KBS-3V variant and their safety functions are described in TKS-2009 (Posiva 2009a, Section 6.1.2). Auxiliary components of the KBS-3H variant and their safety functions will be specified in the Complementary Study Phase in 2010.

2.3.4 Performance targets and target properties

Preliminary performance targets have been defined for the engineered barriers, related to the capacity of these barriers to fulfil their safety functions. Host rock target properties[5], related to the contribution of the rock to the performance of the engineered barriers and to retention of radionuclides within the rock, have also been defined as part of the set of rock suitability criteria (RSC). Because of the favourable features of the main barriers outlined in Chapter 4 (Section 4.1.2), the performance targets are generally expected to be achieved, and target properties are expected to be present. If this is the case then the repository barriers can be assumed to fulfil their respective safety functions. If, however, plausible situations can be identified where the performance targets or target properties are not achieved or present, then the consequences of loss or degraded performance of the corresponding barrier must be evaluated as part of the safety assessment.

Performance targets and target properties are specific to the characteristics of the Olkiluoto site and to the KBS-3 solution and the choice of canister materials. The performance targets and target properties are also based on certain assumptions regarding system evolution and the nature of the perturbing phenomena that may occur over time, and the validity of these assumptions is considered in the safety assessment.

The concept of performance targets is similar to that of “safety function indicator criteria”, as used by SKB in its most recent safety assessment SR-Can (SKB 2006a). The definition of performance targets is also consistent with Finnish regulations.

“Targets based on high quality scientific knowledge and expert judgement shall be specified for the performance of each safety function. In doing so, one shall take into account the potential changes and events affecting the disposal conditions during each assessment period. In an assessment period extending up to several thousands of years, one can assume that the bedrock of the site remains in its current state, taking however account of the changes due to predictable processes, such as land uplift and those due to excavations and disposed waste” (STUK YVL E.5, Section 4.7).

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[5] In the case of the host rock, the term target properties is used in place of performance targets, since the properties of the rock are mainly a function of the selected site and position of the deposition hole, and are relatively insensitive to engineering design.

The preliminary performance targets for the canister are given in Table 2-1.

Preliminary target properties for the host rock and performance targets for the other main barriers are presented in TKS-2009 (Posiva 2009a, Section 6.1.4), and these will be further developed and applied in support of the Preliminary Safety Assessment Report (PSAR) in 2012. The performance targets and target properties are qualitative, but they have been given a quantitative target value where appropriate, based on present-day scientific understanding. The target values may be subject to change in the light of ongoing and future research, technical design and development work.

Table 2-1. *Canister performance targets with possible target values and associated rationale. All targets apply up to several hundred thousand years in the future (after Posiva 2009a, Table 6-1).*

Performance target	Rationale
Canister shall be water and gas tight. Copper shall completely cover the canister interior. Canister shall have good corrosion resistance in the near field chemical conditions that evolve through time.	The isolation of the radionuclides in the spent fuel. Due to radioactive decay, the activities of most radionuclides will have decreased significantly after 100 000 years.
Canister shall withstand the expected isostatic mechanical loads. Canister shall have sufficient mechanical strength to ensure minimal probability of isostatic collapse for isostatic pressures of up to 45 MPa.	The isolation of the radionuclides in the spent fuel (see above). The design isostatic load of 45 MPa is based on a consideration of hydrostatic pressure at repository depth, buffer swelling pressure and the estimated load from a 3 km maximum thick ice sheet. The load required for isostatic collapse will be defined by tests and modelling. The effect of uneven load due to variations in buffer swelling pressure also needs to be examined.
Canister shall withstand the expected dynamic mechanical loads. Canister shall have sufficient mechanical strength to ensure rupture limit > maximum shear stress on the canister, corresponding to a 10 cm displacement with a velocity of 1.0 m/s in any direction in the deposition hole *.	The isolation of the radionuclides in the spent fuel (see above). Dynamic mechanical loads may occur after deglaciation. Loads on the canister caused by minor rock shear movements also depend on the rheological properties and temperature of the buffer. Buffer should not be too stiff.

* This target value is under re-evaluation.

3 DISPOSAL SYSTEM CHARACTERISTICS AND IMPLEMENTATION

This chapter (Fig. 3-1) presents a description of the main components of the disposal system for which a safety case is to be made, including both the KBS-3V and KBS-3H variants, and of how the repository will be implemented.

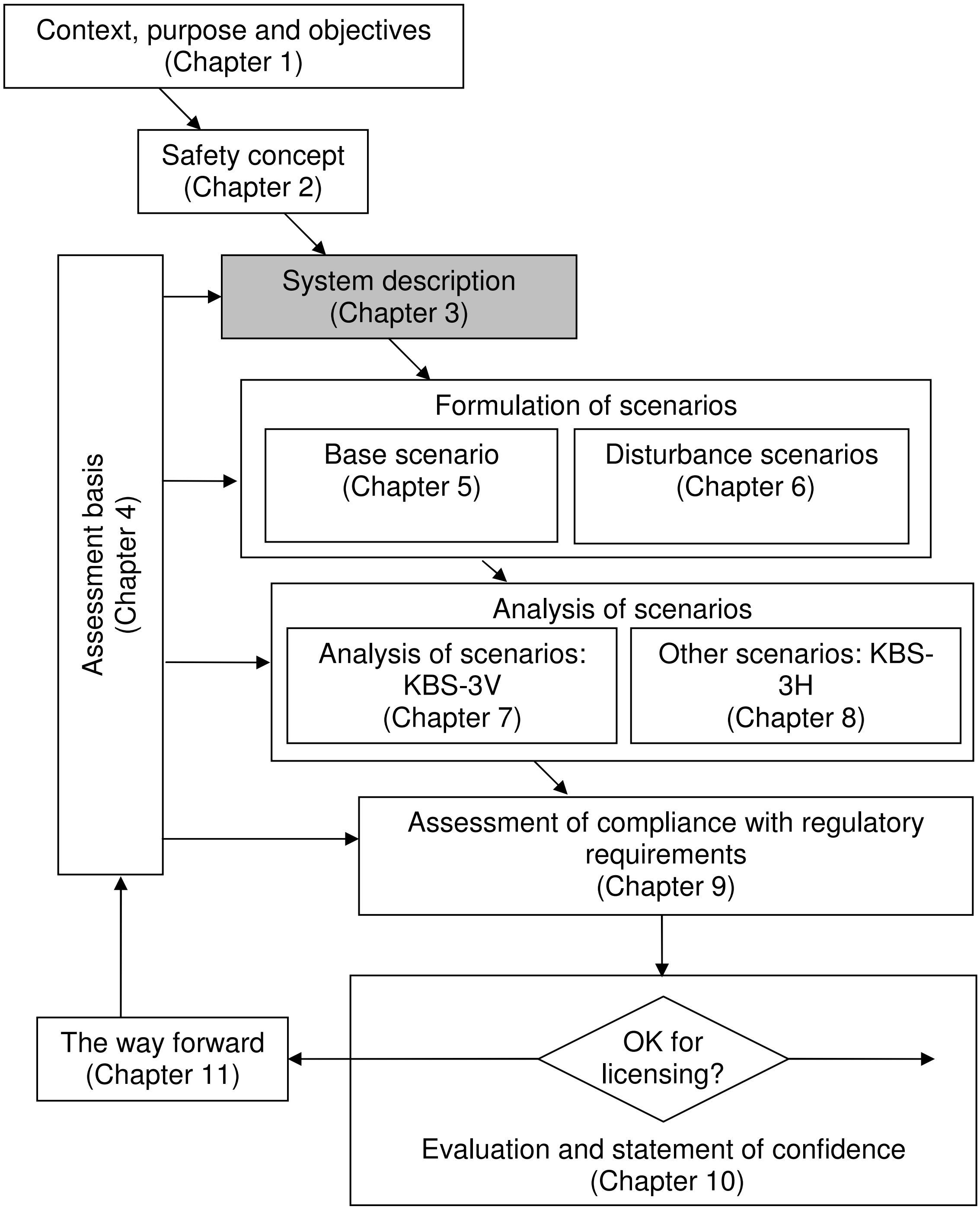


Figure 3-1. *The present chapter in the context of the safety case summary report.*

The disposal system comprises the repository, i.e. the system of engineered barriers and the surrounding bedrock, plus the overlying surface environment. The KBS-3V and KBS-3H variants have largely similar engineered barriers and identical geological and surface environments, and so much of the description applies to both, although important differences are indicated where appropriate.

The chapter is structured as follows:

- Section 3.1 describes the general characteristics, activity and radionuclide inventory of the spent fuel for disposal;
- Section 3.2 summarises the characteristics of the Olkiluoto site;
- Section 3.3 describes the main engineered barriers; and
- Section 3.4 described safety-relevant aspects of repository implementation.

3.1 Spent fuel properties

3.1.1 General characteristics

Several types of spent fuel are planned for disposal in the Olkiluoto repository. These types originate from the Finnish nuclear power plants operated by TVO and Fortum:

- BWR spent fuel from the boiling water reactors at OL1 and OL2;
- PWR spent fuel from the pressurised water reactors LO1 and LO2, which has two sub-types: TVEL VVER-440 and BNFL VVER-400;
- PWR-type spent fuel from the EPR reactor OL3, currently under construction.

The main features of spent fuel of these types are described in the report by Anttila (2005).

Nuclear fuel consists of sintered uranium dioxide (UO_2) pellets enriched with roughly between 3 and 5% U-235. The pellets are stacked, according to each nuclear core's design specifications, into sealed cladding tubes of corrosion-resistant metal alloy (Zircaloy or Zr-Nb alloys). In a nuclear power reactor, the resulting fuel rods are grouped in assemblies that are used to build the nuclear fuel core.

Irradiation of the fuel assemblies produces a large number of radionuclides. These radionuclides include those produced by the fission of uranium and plutonium in the fuel pellets (fission products), as well as activation products arising from neutron absorption. The majority of fission products and higher actinides in spent fuel exist as a solid solution in the uranium dioxide matrix. However, some of the activation products, such as C-14 and Cl-36, are present in both the fuel pellets and in structural materials. Certain radionuclides (e.g. I-129, C-14 and Cl-36) are also enriched at grain boundaries in the fuel, at pellet cracks and in the fuel/sheath gap as a result of thermally driven segregation during irradiation of the fuel in the reactor, as illustrated in Figure 3-2. The radionuclide content of spent fuel assumed in recent Posiva safety assessments is presented in Nykyri et al. (2008) and in Smith et al. (2007c).

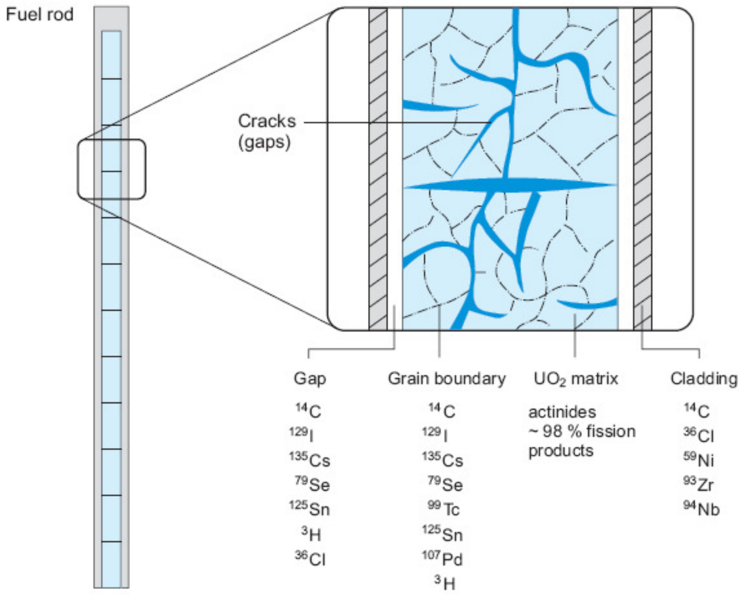


Figure 3-2. *Schematic illustration of the distribution of radionuclides within a fuel rod (from Nagra 2002, based on Johnson & Tait 1997).*

The activity and radionuclide content of spent fuel are functions of its initial enrichment, its burn-up history in the reactor, and the elapsed time since it was removed from the reactor. The minor differences in initial activities arising from the different spent fuel types, and their decay over time after unloading from the reactor (i.e. cooling time), are illustrated in Figure 3-3 for fuel with an initial enrichment of between 3.6% and 4% and a burn-up of 40 MWd/tU.

Burn-up is a measure of the neutron irradiation of the fuel during its time in the reactor and of the energy that the fuel produces. 40 MWd/tU, as assumed in recent safety analyses, is at the high end of the currently expected range. However, in a power station, high fuel burn-up is desirable because it reduces both the amount of downtime needed for refuelling and also the number of fresh nuclear fuel elements required (and spent nuclear fuel elements generated) while producing a given amount of energy. Thus, higher burn-up fuels may also need to be disposed of. Generally, higher burn-up fuel has a higher activity and decay heat on unloading from the reactor. The long-term safety and other implications of higher fuel burn-ups should they be disposed of in the repository are discussed in the Environmental Impact Assessment of 2008 (Posiva 2008a).

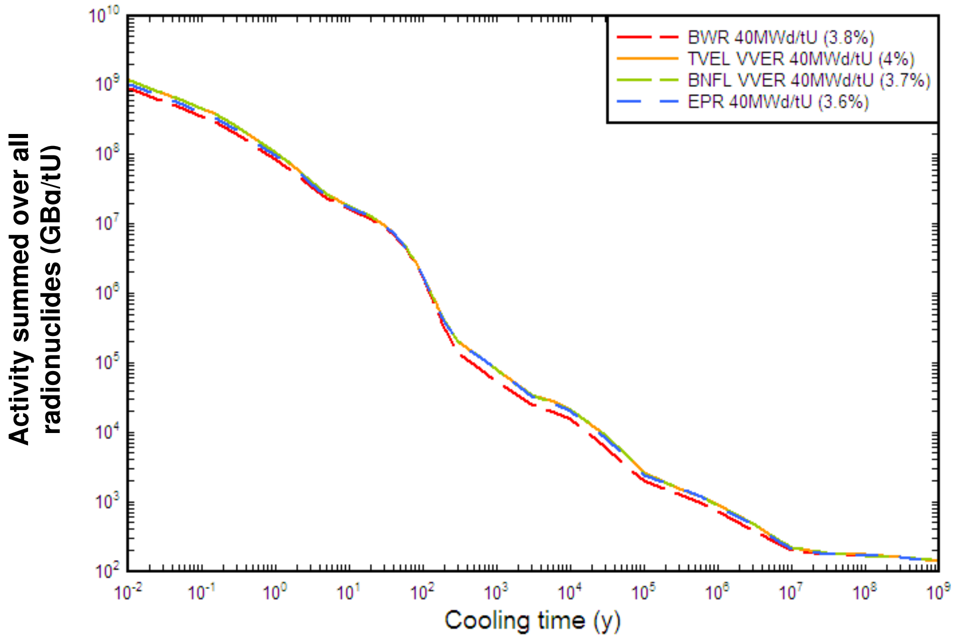


Figure 3-3. *Four different spent fuel types and their radioactivity summed over all radionuclides per tonne of uranium as a function of time after unloading from the reactor (data from Anttila 2005).*

3.1.2 Amounts of spent fuel and radionuclide inventory

The repository layouts considered in recent Posiva safety assessments are based on estimates of the amount of spent nuclear fuel to be produced in each of their reactor units by TVO and Fortum (data from Table 4-1 of TKS-2009, Posiva 2009a):

Loviisa 1-2:	• 641 canisters containing a total of 941 tU of spent fuel
Olkiluoto 1-2	• 1 219 canisters containing a total of 2 555 tU of spent fuel
Olkiluoto 3:	• 960 canisters containing a total of 2 054 tU of spent fuel
Total:	• 2 820 canisters containing a total of 5 550 tU

These estimates are, however, subject to uncertainty and will have to be increased, for example, if the operational lifetime of the reactors is extended. Furthermore, the possible need to accommodate spent fuel from the planned fourth reactor at Olkiluoto and the planned third reactor at Loviisa could require the total capacity of the repository to be up to 12 000 tU (see Section 1.1.1). The long-term safety and other implications of this larger quantity of waste are again discussed in the Environmental Impact Assessment of 2008 (Posiva 2008a).

For the PSAR, the assumptions concerning the amounts of spent fuel, the radionuclide inventory and other spent fuel specifications will be decided on the basis of the status of plans in 2010-2011.

3.2 The Olkiluoto site

Characterisation of the Olkiluoto site has been ongoing for over 20 years, using airborne and ground surveys, shallow and deep (300 to 1 000 m) boreholes and, since 2004, by various measurements taken and investigations made during the construction of the ONKALO underground research facility. The information available before ONKALO construction is summarised in Anttila et al. (1999) and in the Baseline Report (Posiva 2003).

Since the construction of ONKALO, Posiva has compiled a succession of site descriptive model (SDM) reports, presenting an integrated description of the understanding of the Olkiluoto site (Posiva 2005, Andersson et al. 2007 and Posiva 2009b). A description of the current state of the surface environment at Olkiluoto and the main features of its past evolution is integrated within the SDM, and presented in more detail in the Biosphere Site Description BSD-2009 (Haapanen et al. 2009a). BSD-2009 is an update of the Olkiluoto Biosphere Description 2006 (Haapanen et al. 2007). It provides conceptual ecosystem models and assessment data to support biosphere assessment. The planned Models and Data Report of the safety case portfolio will also document data about the site in a manner designed to provide traceability of the data used in safety analyses.

In general there is a high level confidence in the key aspects of Olkiluoto Site descriptive model, especially in the central part of the island based on the relative wealth of the data from this area and the consistency between the different discipline models. There is currently less confidence in the characteristics of the eastern part of the island due to lower amount of data.

The following sections give a brief description of some important safety-related aspects of the Olkiluoto site, based on Chapter 11 of the latest version of the SDM (Posiva 2009b) and on the BSD-2009. The reader is referred to these reports for references supporting the statements given in these following sections.

3.2.1 Geological setting and exploitable natural resources

Olkiluoto is an island approximately 12 km² in area, with an average height of 5 m above sea level, located in the Bothnian Sea off the coast of Southwestern Finland. The gulf occupies a depression in the Fennoscandian Shield. Thus, the bedrock and the landforms are very similar on both the sea bottom and the adjacent land. As a result of post-glacial land uplift, new areas emerge from the sea throughout the coastal areas of Finland, the rate of uplift being 3 to 9 mm per year (about 6 mm per year at Olkiluoto).

The bedrock is divided into following rock types: i) metamorphic rocks, which include various migmatitic gneisses and homogeneous, banded or only weakly migmatised gneisses and ii) igneous rocks, which comprise abundant pegmatitic granites and sporadic narrow diabase dykes. Ductile deformation during the Svecofennian Orogeny ca. 1.91 to 1.80 Ga ago resulted in lithological layering, foliation, strong migmatisation, and thrust-related folding. Subsequently, several tectonic events occurred, resulting in brittle deformation. Hydrothermal alteration processes resulted in mineralogical changes that are, in places, restricted to incipient fractures and to narrow zones adjacent to them. Elsewhere alteration products occur as spots or are finely disseminated throughout the rock and in the fracture fillings. Typical fracture filling materials include clay minerals, calcite and sulphides and their combinations.

The bedrock has no economic potential for hydrocarbon extraction (although there is a notable content of methane in the groundwaters at depth, it does not constitute a commercial resource). The low geothermal gradient makes geothermal energy exploitation unlikely and there is no evidence for metal ores or other industrial mineral deposits locally that might be considered commercially viable in future (Section 4.3.2 of Neall et al. 2007). The exploitation of the bedrock as stone for commercial applications is not likely to be economically feasible in the area, and any quarries are most likely to affect only the shallower parts of the bedrock.

3.2.2 Rock fracturing and groundwater flow

Mechanically less stable and hydrogeologically transmissive brittle deformation zones occur at Olkiluoto as either gently SE-dipping thrust faults or approximately N-S or NE-SW striking strike-slip faults, both types representing reactivated, earlier, non-brittle features. The fault zones are conceptualised as consisting of either a single or several fault core zone(s), with a zone of influence adjacent to the core or cores (see Figure 3-4). The zone of influence show signs of less intense deformation than the fault

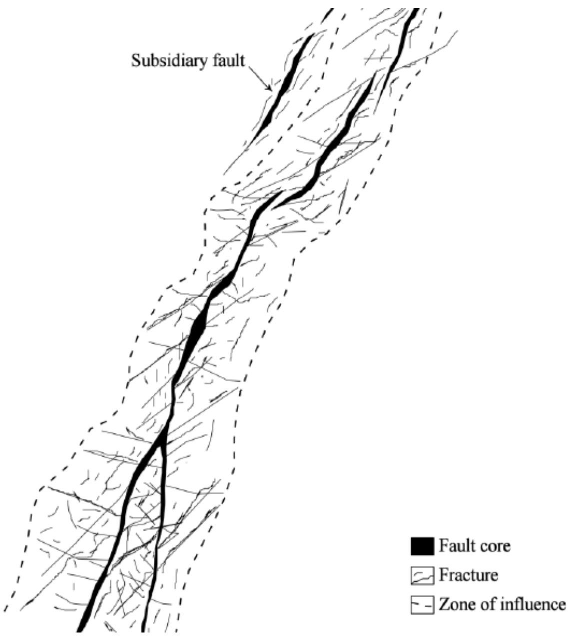


Figure 3-4. *Conceptual model of the fault structure; fault core and zone of influence (Posiva 2009b).*

cores, but is distinguished from the average rock by higher fracture intensity, geophysical anomalies and higher number of hydraulically conductive fractures.

Outside the brittle deformation zones, there is a clear decrease of fracture intensity with depth, the average number of fractures at repository depth being 2 fractures per metre. A statistical description of fracture orientation, intensity and size is provided by a geological Discrete Fracture Network (DFN) model. Three fracture sets are modelled, one sub-horizontal and two sub-vertical. The main uncertainties related to the geological DFN model are related to the fracture size distribution and the variability of the fracture intensity. Different model variants have been produced to calibrate the model against the outcrop, drillhole and tunnel data.

Fractures and fracture zones constitute dominant paths for groundwater flow. Transmissive fractures at repository depth, especially those with transmissivities higher than 10^{-8} m^2 s^{-1}, are concentrated mainly in local zones of abundant fracturing. Fractures with lower transmissivities occur outside these zones. An equivalent porous medium (EPM) model and a discrete fracture network (DFN) model have been developed to describe the distribution of the groundwater flow at the site, including the influence of the planned repository. The hydraulic properties of the fractures (transmissivity distribution) in the hydrogeological DFN are defined based on Posiva flow log (PFL) results. The main uncertainities in the hydrogeological DFN model are related to the fracture size and frequency. Alternative models have been produced to consider different fracture size-transmissivity correlation and treatment of fractures with transmissivity below the PFL measurement limit. The hydrogeological models are described further in Section 4.3.3.

3.2.3 Groundwater composition

The groundwater at Olkiluoto is characterised by a significant range in salinity. Fresh/ brackish waters rich in dissolved carbonate are found at shallow depths, in the uppermost tens of metres. Brackish groundwater, with salinities of up to 10 g/l dominates at depths between 30 m and about 400 m. Saline groundwaters (salinities ranging from about 10 to 84 g/l) dominate at still greater depths. The present-day level of salinity at repository depth is 10 to 20 g/l. The salinity may increase due to possible upconing of saline groundwater during repository construction and operation and in the early post-closure period, but is expected to remain below the 70 g/l, which is specified as a maximum target value for groundwater salinity at repository depth (higher salinities could result in too low swelling pressures in the backfill or buffer, see Posiva 2009a, Section 6.1.4).

Two natural metastable geochemical interfaces are present in the Olkiluoto bedrock. The upper is located in the overburden, where the conditions change from oxic to anoxic. The presence of this interface is supported, in part, by the observed scarcity of iron oxyhydroxides on fracture surfaces beneath the uppermost few tens of meters. Pyrite and other iron sulphides are, on the other hand, common in fractures throughout the investigated depths, indicating a strong lithological buffer against oxic waters over geological timescales. The other interface is located at a depth of approximately 250 to 350 m. In this zone, SO_4-rich groundwater is mixed with dissolved methane, which results, in places, in exceptionally high levels of dissolved sulphide as a microbially mediated reaction product.

The nitrogen and methane gas content is high in Olkiluoto groundwater, nitrogen being the dominant dissolved gas in the upper 300 m with methane being dominant at greater depths. The pH conditions in the deep aquifer system at Olkiluoto are well buffered by the presence of abundant carbonate and clay minerals found in fracture fillings. The pH values at relevant depths are generally in the range 7.5 - 8.2.

3.2.4 Thermal and mechanical properties

The thermal properties investigated at Olkiluoto include: thermal conductivity, specific heat capacity, thermal diffusivity and thermal expansion coefficient. The main focus of the thermal property studies has been the veined gneiss, the main rock type at repository level, a significant characteristic of which is its thermal anisotropy. Thermal conductivity ranges from 2.3 to 3.2 W/mK.

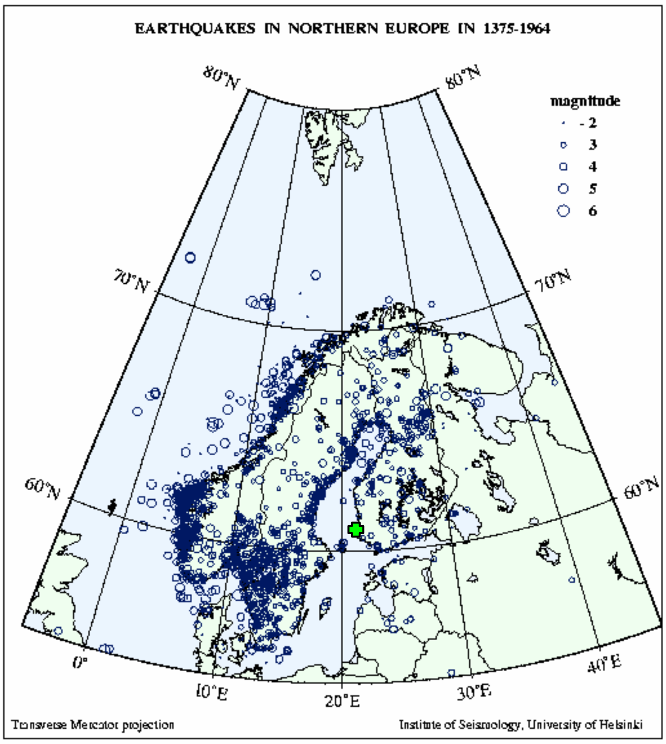
Estimates of rock stress indicate the maximum horizontal component at repository depth at 500 m to be between 15 and 31 MPa, with a generally E-W orientation, though with a relatively large scatter. The minimum horizontal stress is estimated to be in the range 10 to 18 MPa. The vertical stress is estimated to be between 7 and 15 MPa. The major principal stress is subhorizontally orientated, and is thus slightly larger in magnitude than the maximum horizontal stress.

The strength and deformation properties of the intact rock depend essentially on its mineral composition and structure. The variation in the mineral composition of the rock typical to Olkiluoto is reflected in the notable spread of rock strength in the metamorphic rocks. The mean uniaxial compressive strength (peak strength) is around 115 MPa. All the main rock types at Olkiluoto have about the same peak strength.

3.2.5 Seismic activity

The Olkiluoto site, like Finland in general, has low seismic activity, with Global Positioning System (GPS) and seismic measurements at Olkiluoto showing negligible rock movements (see, e.g. La Pointe & Hermanson 2002; Enescu et al. 2003; Saari 2006; Andersson et al. 2007). Earthquakes in Northern Europe since 1375 are shown in Figure 3-5. The figure shows that the density and magnitude of earthquakes in Finland is lower than in many other areas. Earthquake magnitudes have never exceeded 5 on the Richter scale since instrumental records began in the 1880s (Marcos et al. 2007 and references therein).

In the past, in Northern Finland (though not in Southwestern Finland, where Olkiluoto is located) the greatest seismic activity has occurred following the retreat of the ice sheets that covered the region during ice ages. Any large earthquakes in the future are also most likely to occur following glaciations, although infrequent but significant seismic events during inter-glacial periods cannot be excluded. Studies, including modelling of fault reactivation due to advance and retreat of continental glaciers, are underway to better understand the potential for large earthquakes in the future, as described in TKS-2009 (Posiva 2009a, Section 6.5.7.2).



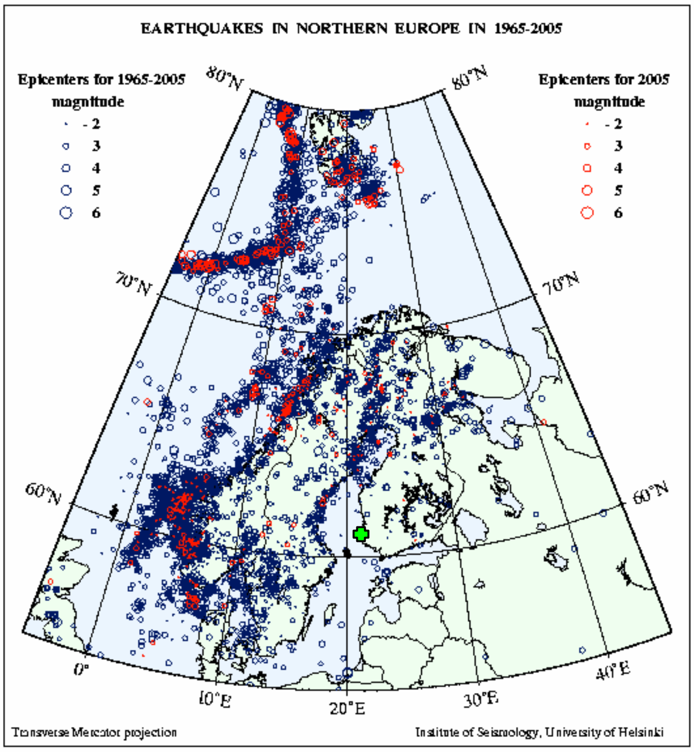


Figure 3-5. *Earthquakes in Northern Europe in 1375-1964 (above) and in 1965-2005 (below). A green cross indicates the approximate position of the Olkiluoto site.*
(http: //www.seismo.Helsinki.fi/bulletin/list/catalog/histomap.html (above)
http: //www.seismo.Helsinki.fi/bulletin/list/catalog/instrumap.html (below))

3.2.6 The surface environment

Olkiluoto Island has a moist climate, short growing season and wintertime snow cover. The annual mean temperatures range from 4 to over 6 ^{o}C and annual precipitation rates range from 410 to 670 mm. Due to the climate, soil types and also low population pressure, the landscape is dominated by forests and mires.

The overburden at Olkiluoto, which is mainly fine-textured and sandy till, is generally 2-5 m thick, although locally thicker soil layers (up to 14 m thick) have been found. The mammalian fauna on the island is typical of coastal areas in Southwestern Finland. Olkiluoto is generally not very favourable for most amphibians and reptiles. Invertebrates found on the island are common in Southern Finland and the land bird population is typical of Southern Finnish coastal forest areas. The area is not important for the occurrence of rare species.

The waters around Olkiluoto Island are shallow except for a few areas where sea depths reach about 15 m. Seashores show a great variation in environmental conditions. Deep hard and soft sand bottoms, as well as shallow bottoms with mostly soft clay, mud and silt are found. The nutrient concentrations in the water in the sea area off Olkiluoto are typical of the coastal waters of the Bothnian Sea. The aquatic flora in the Olkiluoto offshore zone ranges from the algae-dominated hard bottom communities in the outer archipelago to vascular plant-dominated soft-bottom communities in sheltered locations. The most distinctive species in the littoral zone between the high water mark, which is rarely inundated, and shoreline areas that are permanently submerged is the common reed. This forms an almost continuous belt around the island. The varied habitats near the shoreline maintain a great variety of fauna, waterfowl being the most distinctive group.

There are two major rivers, Eurajoki and Lapinjoki, which drain into the Bothnian Sea near Olkiluoto. The catchment of the larger river, Eurajoki, includes the largest lake in Southwestern Finland. The river is a medium-sized to large clay-region river, regulated by dams. The amount of nutrient load varies depending on the discharge and weather conditions, especially precipitation. The fish species present vary from place to place, the most common being perch and roach.

Agriculture is currently of minor importance on Olkiluoto Island. In the wider region (the Eurajoki municipality of Southern Satakunta), cereal production is the dominant form of agriculture, barley and oats being the main species. Other forms of cultivation include production of malt barley, pea, potatoes, sugar beet and oil plants (turnip rape, rape, sunflower). Milk production is important in Southern Satakunta, but not in the Eurajoki municipality. Fields are often located along the rivers and the majority have been drained. Irrigation of crops is rare, and the irrigation amounts are small. Animal production tends to use less fertile fields than special crops and horticulture.

The groundwater table at Olkiluoto follows the form of the surface topography and is mainly zero to 2 m below surface. Olkiluoto Island forms its own hydrological unit; the surface waters flow directly into the sea (Lahdenperä et al. 2005, Posiva 2005). Infiltration of surface water is currently being investigated. Current (and provisional) estimates are approximately 1 to 2% of the annual precipitation. Current indications are that the properties of the overburden are such that higher rates of precipitation will result predominantly in greater runoff and evapotranspiration, rather than higher rates of infiltration to the bedrock.

Sampling of shallow groundwaters indicates that they are mainly fresh and locally slightly brackish, with the corresponding water types being Ca-HCO_3 and Na-Cl. The pH of the samples varies from 5.1 to 8.0. Fresh groundwater is found only at shallow depths, in the uppermost tens of metres. Most of the oxygen carried by precipitation is consumed in the humus layer, and much of the remainder in the mineral soil layers.

The evolution of surface conditions and the local ecosystem are described in Section 5.2.1 of this report and, in more detail, in the Terrain and Ecosystems Development Model Report (Ikonen 2007, Ikonen et al. 2010).

3.3 The engineered barriers

The following sections give a brief description of the main engineered barriers for KBS-3V and KBS-3H, based mainly on the report by Raiko (2005), which describes canister design, dimensioning analyses, manufacture, fuel encapsulation, sealing, quality assurance and quality control, and on the repository layout and design reports for KBS-3V (Saanio et al. 2006, Kirkkomäki 2007) and for KBS-3H (Johansson et al. 2007; Autio et al. 2008). The first version of the Models and Data Report (*Models and Data Report 2010*) of the safety case portfolio will also document data about the engineered barriers in a manner designed to provide traceability of the data used in the recent safety analyses. The reader is referred to those reports for references supporting the statements given in the following sections.

3.3.1 The canister

The canisters that will hold the spent fuel assemblies each consist of a massive cast iron insert covered by a 50 mm-thick copper overpack. Copper has been chosen as the overpack material because of its well-known properties, its favourable thermal and mechanical properties and for its resistance to corrosion in reducing environments. Cast iron has been chosen for the insert to provide mechanical strength, radiation shielding and to maintain the fuel assemblies in the required configuration. There are three versions of the canister, one for each spent fuel type (Figure 3-6). All have the same outer diameter of 1.05 m. The height ranges from 3.6 to 5.25 m. Posiva plans to seal the canister lids using electron beam welding, with friction stir welding as an alternative option. There has been extensive testing of both these welding techniques (see Ch. 5 in Posiva 2009a).

Figure 3-6. *Copper-iron canisters, from left to right, for spent fuel from (i), the LO1 and LO2 (VVER-440) reactors, (ii), the OL1 and OL2 (BWR) reactors and (iii), the OL3 (EPR/PWR) reactor (after Raiko 2005).*

3.3.2 The buffer

Bentonite provides the buffer material that surrounds the canisters and separates them physically from the rock. The major component of bentonite is montmorillonite, which confers its swelling properties. Other components are quartz, feldspar, cristobalite, gypsum, calcite and pyrite. These components can play an important role in determining buffer pore water composition. MX-80 sodium bentonite has been the reference buffer material in Posiva's safety assessments to date. The properties of MX-80 are well known, based on extensive and thorough national and international studies. However, another potential buffer material, calcium bentonite DEPONIT CA-N, is also currently under consideration. The buffer around the canisters will be installed as partly saturated rings and blocks. The initial density of the rings and blocks will be based on a design density following saturation of 2 000 kg/m^3.

3.3.3 The KBS-3V deposition tunnel backfill

The KBS-3V deposition tunnels will also be backfilled with a swelling material to limit water flow near the deposition holes, prevent the loss of buffer density and provide mechanical stability to the tunnels. Posiva is currently developing a site-specific backfilling concept within the Coordination programme for Olkiluoto-specific backfill (OBA). Olkiluoto-specific issues related to design and long-term safety will be identified, and backfill materials selected. The backfilling methods under consideration are installation of pre-compacted blocks and bentonite pellets and in situ compaction, of which the block backfill is the considered as the reference concept. Three different materials are being considered for block materials:

- Friedland clay, which is the current reference block material, with an estimated maximum swelling minerals content of 30%;

- Milos B bentonite clay with an estimated amount of swelling minerals of 50 - 60%; and
- a mixture of bentonite and ballast with a minimum bentonite content of 40% (of which over 70% is composed of swelling minerals).

A backfill design report will be published in early 2010. Furthermore, a description of the backfill properties assumed in recent Posiva safety assessments, including physico-chemical information, volumes and masses as well as their location in the repository, will be given in the planned Models and Data Report (Fig. 1-3).

3.4 Repository implementation

3.4.1 Layout and rock suitability criteria

A provisional layout for a repository at Olkiluoto is given in Figure 3-7. The layout is determined by considerations of both engineering feasibility and long-term safety. The layout is, for example, adapted according to:

- the distribution of deformation zones with unfavourable mechanical and hydrogeological properties, or with the potential to host large earthquakes;
- the thermal properties of the rock, such that excessively high temperatures at the canister surface and in the surrounding buffer are avoided; and
- the direction of the maximal horizontal stress for reasons of mechanical stability.

The respect distance to major deformation zones is based on the influence zone that exists around them. The influence zone, often called damaged zone, next to the fault core, which is characterised, for example, by increased fracturing, alteration, or the presence of geophysical anomalies.

Repository layout will, in future, also be guided by the ongoing work of the Rock Suitability Criteria (RSC) programme, which has been set up to provide criteria for locating suitable rock volumes for repository construction and, at a smaller scale, for canister deposition. The RSC will be used to identify volumes of host rock with the target properties, and where it is also likely that the performance targets set for the canister, buffer and backfill will be achieved. Tentative target properties and performance targets are given in TKS-2009 (Posiva 2009a, Section 6.1.4).

One key parameter for which RSC are under consideration is initial groundwater inflow and its impact on the buffer. Key aims will be (i), to avoid canister and buffer emplacement in positions where piping and erosion may be a problem, (ii), to limit the potential for changes in chemical conditions in the deposition holes or drifts and (iii), to limit radionuclide transport in the event of canister failure. Another key issue being addressed by the RSC programme is that larger fractures, if not avoided at canister emplacement locations, may undergo significant secondary shear movements in the event of a large earthquake that could damage the canisters (Section 6.1.3). Although the RSC are still under development, it is not envisaged that their application will give rise to any problems as to the total volume of rock needed for emplacement.

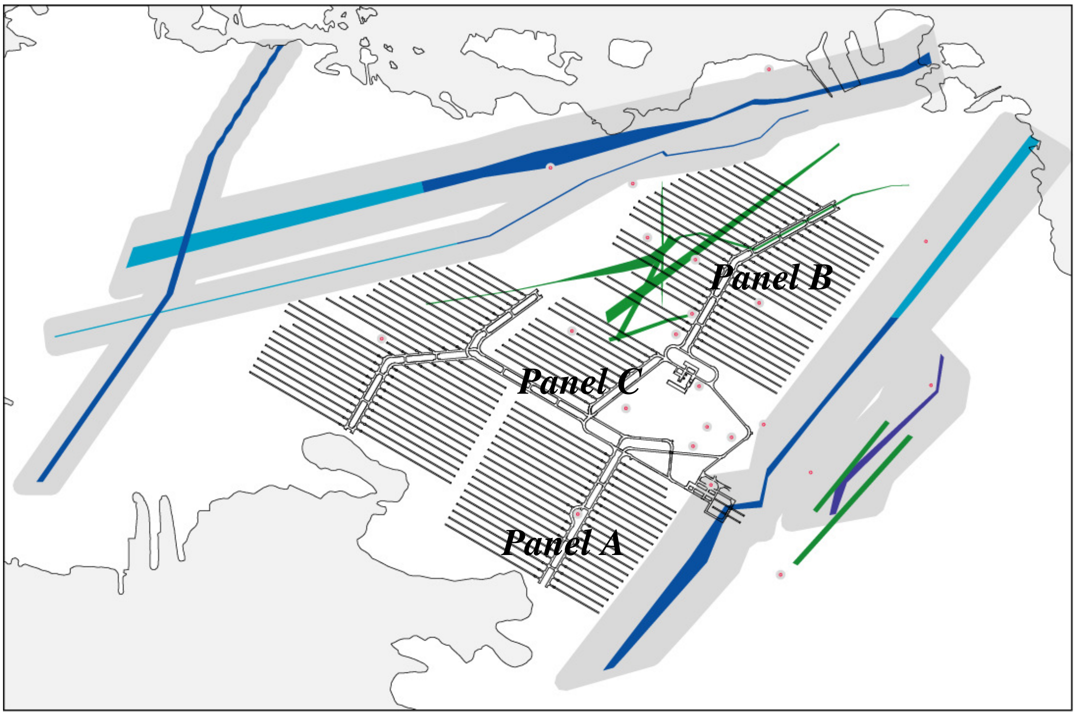


Figure 3-7: *Example layout for a KBS-3V type repository at Olkiluoto (after Kirkkomäki 2007). Blue features determine the repository layout. Green features are avoided when locating individual deposition holes. A provisional layout adapted to the KBS-3H repository, broadly similar to that for KBS-3V, is given in Johansson et al. (2007). The designation of three panels as A, B and C is used in the description of radionuclide transport modelling in the biosphere (Section 7.1.2).*

3.4.2 Use of cement and other construction materials

Significant amounts of cement are being used in the development of ONKALO and cement will also be used, for example, in the KBS-3H compartment plugs and to fill e.g. anchoring holes or grooves or as a grouting material. Detailed descriptions and inventories of cement and other construction materials are presented in Hagros (2007a and 2007b). Once operations in a KBS-3V deposition tunnel or KBS-3H deposition drift are complete, a plug will be emplaced at its open end to avoid the possibility of buffer erosion by water flowing out of the tunnel, either through the plug itself, or through the adjacent rock, and to keep the backfill in place prior to the backfilling of the central tunnels. The plug will also contain significant amounts of cement according to the current design.

It is currently foreseen that low-pH cement and/or Silica Sol (colloidal silica) grouts[6] will be used for grouting fractures and for any other components in close proximity to the buffer and canisters. These materials are expected to have reduced detrimental

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6 Colloidal silica is being considered as a grouting material for narrow fractures (Autio et al. 2007). Colloidal silica is a stable dispersion of discrete nonporous particles of amorphous silicon dioxide (SiO_2). The long-term durability and evolution of colloidal silica grouts has not been as well characterised as that of cementitious grouts (Ahokas et al. 2006). For example, the potential for forming colloids with radionuclides is under investigation.

effects, most notably on the buffer, compared with Ordinary Portland Cement (OPC), because of the reduced alkalinity of their porewater leachates (see Posiva 2009a, Section 6.5.4.2). The durability and evolution of such materials is not currently as well understood as for the OPC. However, durability of components made from these materials is only required for the operational period and not in the longer term. Low-pH cement, silica sol and OPC will continue to be studied in Posiva's research, technical design and development programme, including sensitivity analyses to test the robustness of the KBS-3 disposal concept against (hyper)alkaline attack (Posiva 2009a, Section 6.5.4.2).

Where practicable, the introduction of potentially detrimental stray materials into the repository will be avoided, or such materials will be removed during repository operations. Some residual materials will, however, inevitably remain in the repository and the surrounding rock after closure. Potential amounts have been estimated, based on the current reference layouts for KBS-3V and KBS-3H (Hagros 2007a, b). Residual material may include organic substances. The use of low-pH cements will also necessitate the use of organic additives such as superplasticisers to maintain workability and other physical properties. Posiva, SKB, Nagra and NUMO have conducted a joint project: Superplasticisers and Other Organic Cement/Concrete Admixtures: Long-term Safety Aspects. The aim of this project was to develop a methodology for the evaluation of the long-term safety implications of the use of superplasticisers (SP) and other organic additives used in cement, and to evaluate the effects of SPs and other additives that have already been used or that are most likely to be used in the construction of high-level nuclear waste repositories. The results are published in Andersson et al. (2009).

3.4.3 The possibility of deviations from design, accidents or mishaps

Repository excavation, the handling and transport of canisters from the encapsulation plant to a KBS-3V deposition hole, KBS-3V buffer and backfill emplacement and assembly and emplacement a KBS-3H supercontainers, distance blocks and other components will all be performed according to prescribed quality control procedures. Nevertheless, unexpected deviations from design, accidents or mishaps during construction and operations cannot be completely excluded. Nevertheless, in the case of the canisters, the presence of non-detected penetrating defects or other defects that could lead to early canister failure cannot currently be excluded. Similarly, the possibility of improper buffer emplacement cannot currently be excluded. These possibilities are thus taken into account in the safety analyses (see Section 6.2.1).

3.4.4 Backfilling of remaining openings and repository closure

Posiva is currently developing backfilling concepts for the other excavated parts of the repository (central tunnels, auxiliary rooms, access tunnels, shafts) as well as methods for the final closure of the repository, such as plugging and sealing methods. Repository closure and sealing plans are common for KBS-3V and KBS-3H. The backfilling and sealing concepts have been developed together with SKB within the Baclo (Backfilling and closure of the repository) programme. Reports on backfill methods and plans have been published in collaboration with SKB (Gunnarsson et al. 2007, Keto & Rönnqvist

2006; Keto et al. 2009). Current plans for plugging and sealing are described in Posiva (2009a), Section 6.5.6.1.

4 ASSESSMENT BASIS AND METHODOLOGY

This chapter (Fig. 4-1) presents a description of the assessment basis, which is the scientific and technological information, understanding, models and data on which the safety assessment and safety case are based, and the methodology for carrying out the safety assessment.

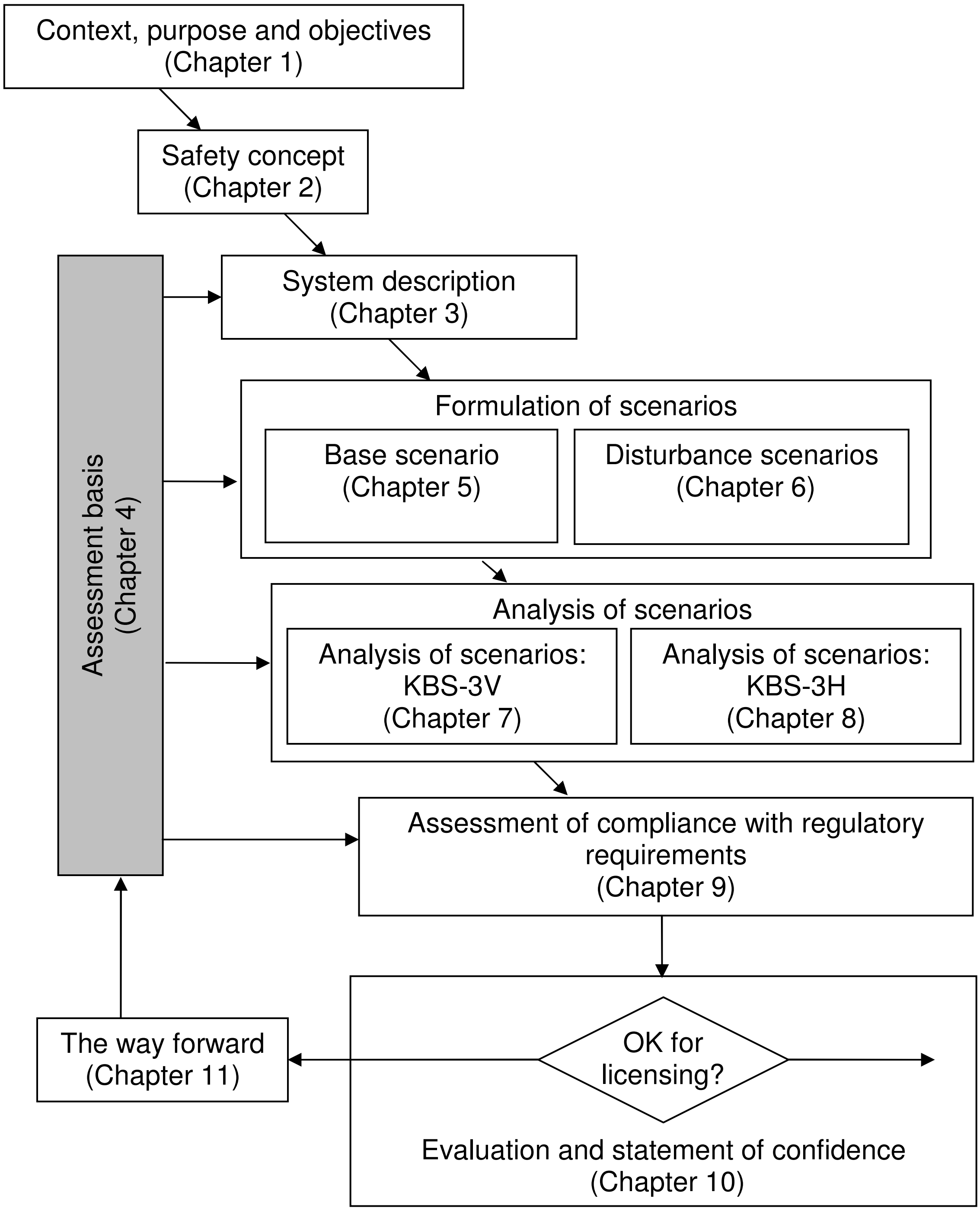

Figure 4-1. *The present chapter in the context of the safety case summary report.*

The chapter is structured as follows:

- Section 4.1 discusses the features, events and processes (FEPs) relevant to safety assessment and to the safety case, including key features of the main barriers contributing to the safety functions, potentially detrimental features, events and processes, and how these are checked for completeness;
- Section 4.2 describes how scenarios for the evolution of the disposal system over time are classified and formulated;
- Section 4.3 discusses the analysis of scenarios by means of modelling;
- Section 4.4 deals with radiological consequence analysis and safety indicators;
- Section 4.5 describes the computer codes used in safety assessment.

4.1 Features, events and processes

4.1.1 Identification and documentation of features, events and processes

Theoretical or conceptual understanding of relevant features, events and processes (FEPs) and their interactions forms the basis for assessing the long-term safety of the proposed repository and producing a safety case. Posiva has not, to date, compiled its own FEP database and report, although planning for these is underway for 2010. The current understanding of FEPs is documented, to a large extent, in various versions of the Process Report, which is a main report in the current safety case portfolio (Fig. 1-3) documenting process understanding, and the safety relevance of processes in different time windows. The planned FEP database report will supplement the Process Report, providing, in particular, a description of system features in addition to events and processes. The planned FEP database is seen as a valuable tool for reducing the possibility of omissions in the FEPs considered in scenario formulation and analysis.

The latest version of the Process Report for the KBS-3V variant was produced in 2007 (Miller & Marcos 2007). A separate KBS-3H Process Report was jointly issued by Posiva and SKB (Gribi et al. 2007). The KBS-3H Process Report employed a "difference analysis" approach whereby the main focus was on processes that have a different significance for, or potential impact on, the safety functions and evolution of a KBS-3H repository compared with KBS-3V. Biosphere-specific processes are included in neither of these earlier reports. However, a number of processes relevant to the biosphere are discussed in the Biosphere Site Description (Haapanen et al. 2007, 2009a), in the reports on terrain and ecosystem development modelling (Ikonen 2007, Ikonen et al. 2010), and in the reports on landscape modelling (e.g. Broed et al. 2007, Hjerpe et al. 2010). Biosphere processes will be included in the next update of the Process Report, and described in detail in various background reports.

4.1.2 Key features of the main barriers contributing to the safety functions

As described in the previous sections, long lifetime of the canister is central to the safety concept. Features that favour a long canister lifetime are its mechanical strength and its corrosion resistance.

The mechanical strength of the canister is provided mainly by its cast iron insert. Numerical studies have shown the minimum collapse pressure of the canister to be 80-114 MPa, depending on insert type (Ikonen 2005a). SKB has also carried out laboratory pressure tests of full-size inserts with copper overpacks, giving a measured pressure for general collapse of the canister of 138 MPa (Nilsson et al. 2005). Dillström (2005) found that, although some local collapse may occur at lower pressures, the probability of local collapse is insignificant ($\sim 2 \times 10^{-9}$) for a base case pressure of 44 MPa. These results are discussed further in Section 6.1.2 of Pastina & Hellä 2006. 44 MPa is significantly higher than the 11-12 MPa isostatic load exerted on the canisters in the absence of ice cover at the site (Section 8.1.2 of Pastina & Hellä 2006; Section 5.4.3 of Smith et al. 2007b). The presence of a 3 km thick ice sheet will increase the load on the canister by about 27 MPa, but the likelihood of local collapse is expected to remain low (Section 8.1.2 of Pastina & Hellä 2006; Section 7.4.4 of Smith et al. 2007b)[7].

Corrosion resistance is provided by the copper overpack in which the cast iron insert is placed. Copper exhibits excellent corrosion properties under the expected repository conditions, corroding uniformly with little tendency to localised corrosion or stress corrosion cracking (King et al. 2002, 2010). Under oxic conditions, the maximum amount of corrosion can be estimated based on the total amount of oxygen trapped in the repository at the time of closure, amounting to some tens to hundreds of microns if uniformly distributed over the surface of the overpack. Once that oxygen, and the cupric species that can form due to the oxidation of copper by O_2, have been consumed, corrosion will cease unless sulphide is present. In the presence of sulphide, corrosion can occur with the evolution of H_2 and proceeds at a rate determined by the rate of diffusion of sulphide to the surface of the overpack. Because of the presence of chloride ions in the bentonite pore water, the overpack will corrode "actively" with little or no localised penetration. In addition, due to the absence of specific aggressive chemical reagents, i.e., ammonia, nitrite, or acetate, as well as the inhibiting effect of chloride, the overpack will not be subject to stress corrosion cracking. Scoping calculations for a KBS-3V repository presented in Section 11.3.1 of Pastina & Hellä (2006) and for a KBS-3H repository in Appendix B of Smith et al. (2007b) indicate that, if the buffer performs as expected, several hundreds of thousand of years or more will elapse before the overpack fails.

The bentonite clay buffer that surrounds the canister provides it with both physical and chemical protection. Key features of the clay are its low hydraulic conductivity, its swelling pressure, its chemical buffering capacity, its sorption potential and its plasticity. Because of the low hydraulic conductivity of the buffer once saturated, migration of corrosive agents from the groundwater to the canister surface will occur only slowly, predominantly by molecular diffusion (the bulk hydraulic conductivity of the buffer at its design saturated density will be less than 10^{-13} m s^{-1}, see Fig. 4-8 of SKB 2006a). Diffusion-dominated transport and sorption reactions will also strongly limit the spatial extent of any mineralogical changes to the buffer due to reaction with species

[7] Manufacturing tolerances lead to differences which have an impact on the mechanical strength of the canister, in particular in light of its behaviour in future glacial conditions. These can be significant from a long-term safety point of view and will be evaluated in forthcoming studies (Posiva 2009a).

present at the buffer/rock interface, such as interaction with leachates from cementitious components and iron-bentonite interaction processes. The bentonite will exert a swelling pressure on the deposition hole or deposition drift walls that will ensure a tight contact, ensuring slow, diffusion-dominated transport across the buffer/rock interface. Microbial activity in the buffer that could potentially lead to detrimental chemical conditions at the canister surface will be negligible at the design saturated density and swelling pressure (Stroes-Gascoyne et al. 2006; Masurat 2006). The chemical buffering capacity of the clay will ensure that pH conditions at the canister surface are relatively unaffected by, for example, any high pH-plume originating from cementitious repository components (even if exposure were to occur, it is likely to have a favourable impact on corrosion of the canister surfaces, through the formation of a passivating film, see King 2002). The plasticity of the clay will protect the canister from small rock movements, including shear movements on fracture planes due to rock excavation and heat load during early evolution, or due to seismic activity (Sections 7.4.1 and 8.1.2 of Pastina & Hellä 2006 and Sections 5.4.6 and 7.4.4 of Smith et al. 2007b). The plasticity and swelling pressure of the clay will also ensure self-sealing of any potential advective pathways for flow and transport that may arise as a result of any such rock movements, or other processes including the release of gas formed in a damaged canister. The stiffness of the clay is such that canister sinking is expected to be negligible (Börgesson & Hernelind 2006a). The KBS-3V tunnel backfill is chosen to have a low compressibility that will keep the buffer in place in the deposition hole, maintaining its density and swelling pressure (Section 6.2.2 of Pastina & Hellä 2006).

Favourable features of the bedrock at the Olkiluoto site include its low seismic activity (Section 3.2.5), its lack of exploitable resources that might lead to inadvertent human intrusion (Section 3.2.1), its sparse fracturing and the low groundwater flow rates at repository depth (Section 3.2.2) and geochemical conditions that are broadly favourable to backfill, buffer and canister longevity and performance. In particular, canister longevity is favoured by the anoxic conditions that currently exist at repository depth, and are expected to be maintained in the future because of a strong mineralogical buffer against the future penetration of oxic waters indicated by the presence of pyrite and other iron sulphides in fractures (Section 3.2.3). Sparse fracturing and low groundwater flow also limit the supply of other corrosive agents (especially sulphide) from the groundwater to the canister surface.

Although the rock is spatially variable, rock suitability criteria (RSC) will be used to ensure that deposition holes or tunnels are located at positions that provide a protective environment for the canisters and buffer and backfill and limit the radionuclide transport in the event of canister failure (Section 3.4.1). Fractures will, for example, be avoided if they are likely to undergo significant secondary rock movements in the event of a large earthquake (Section 6.1.3).

A range of processes will operate that will severely limit the release of radionuclides to the surface environment in the event of canister failure. The fuel matrix incorporates the majority of the radionuclide inventory and is resistant to degradation by water entering the canister in the expected chemical environment. The fuel assemblies are also manufactured from highly corrosion-resistant materials such as stainless steel, Zircaloy, Inconel and Incoloy. Thus, release rates of radionuclides incorporated within these materials will be slow. In the case of the fuel matrix, recent experimental results of Grambow et al. (2008) support the low fuel dissolution rate values used in the safety analyses described in the present report (a fractional dissolution rate of 10^{-6} to 10^{-8} per year, based on the recommendations of Werme et al. 2004). In reality, the rate of dissolution or alteration of the fuel matrix may be inhibited over time by the build-up of secondary phase deposits on the fuel surfaces, limiting the amount of water contacting the unaltered fuel. This process has not, however, been quantitatively evaluated. Transport of radionuclides from the canister interior, through the buffer and geosphere is expected to be slow. Slow transport combined with radioactive decay will strongly delay and attenuate any releases to the surface environment. The chemical environment within and around the repository, and especially the reducing conditions, will favour low solubility and high sorption of many safety-relevant radionuclides. Significant sorption is expected on the corrosion products of the canister insert and on buffer pore surfaces (although the former is conservatively omitted in radionuclide transport calculations). Radionuclides may become associated with colloids or microbes within and around the canisters, but the saturated buffer has a sufficiently fine pore structure such that microbes and colloids are immobile (filtered) (Section 2.5.4 in SKB 2006b). Radionuclides may also become associated with bentonite colloids formed at the buffer-rock interface and transported into rock. This is an issue still under study (Posiva 2009a, Section 6.5.7.3). Groundwater flowing in fractures will convey released radionuclides through the bedrock. However, the RSC will favour low groundwater flow in the vicinity of the canisters. Furthermore, many radionuclides will be strongly retarded by matrix diffusion and sorption on rock matrix pore surfaces, and will decay to insignificant concentrations during transport. Evidence for connected matrix porosity and matrix diffusion in rock samples from Olkiluoto has been reported for example, in Siitari-Kauppi et al. (1995). The most important transport processes are described further in Chapter 4, and quantitatively evaluated in Chapters 7 and 8.

4.1.3 Potentially detrimental features, events and processes

In spite of the favourable features outlined above, features, events and processes that could potentially lead to canister failure, or degrade the capacity of the repository to limit radionuclide transport in the event of canister failure, cannot be excluded. Based on the Process Reports for KBS-3V (Miller & Marcos 2007) and KBS-3H (Gribi et al. 2007), the following potentially detrimental features and processes arising internally within the repository have been identified:

- the possible presence of penetrating and non-penetrating defects in the canisters or other defects that could lead to early releases;
- processes leading to missing, loss or redistribution of buffer mass, with consequences for copper corrosion and radionuclide transport;
- processes leading to perturbation of the buffer / rock interface, also with consequences for copper corrosion and radionuclide transport;
- gas generated internally within the canister; and
- criticality.

In addition, the following perturbing processes arising from events external to the repository have been identified:

- buffer freezing;
- canister failure due to isostatic load;
- migration of oxygen to repository depth;
- loss of buffer due to exposure to glacial meltwater; and
- canister failure due to rock shear.

These uncertain features, events and processes and their potential consequences are described further in Chapter 6 (Section 6.2.1 and Section 6.2.2).

Both inadvertent and deliberate human intrusion could also compromise the repository safety functions. Posiva holds the view that it is exclusively inadvertent human intrusion that falls within the scope of the safety case. This view is consistent with international practice (see, e.g. Position Statement 6 of the German Working Group on Scenario Development: Internationale Zeitschrift für Kernenergie 2008). Inadvertent human intrusion is taken to be any intrusion that occurs not knowing that the repository is present, including the case when the intruder knows that an artificial structure is present underground, but does not know that it contains radioactive waste (it should be noted that the Olkiluoto site has few resources that might attract deep drilling activities that could cause disturbance or damage, see Section 3.2.1). Recovery and exploitation of the spent fuel and high-quality copper in the repository, or deliberate intrusion into the repository for other reasons, cannot be prevented, and may be necessary to future society, but responsibility for potential consequences must lie with the intruder (Grimwood & Thegerström 1990; NEA 1995).

These various uncertain features, events and processes are taken into account in the definition and formulation of scenarios, as described below. There are also certain other potentially detrimental FEPs that have been identified, but are currently considered to lie outside the scope of Posiva's safety case. These include highly unlikely events and events with direct detrimental consequences that greatly outweigh any releases from the repository that they might cause. Meteorite impact falls into this category.

4.1.4 Completeness checking

Posiva's planned FEP database will be used for completeness checking at each stage in the development of the PSAR and the FSAR. In the work carried out to date, completeness checks have also been made by cross-checking specific products against relevant FEP lists and process tables. In particular, in the 2007 Process Report for the KBS-3V variant (Miller & Marcos 2007), the possibility of significant processes having been overlooked was reduced by cross-checking for completeness against:

- the NEA international FEP database (NEA 2000);
- the processes included in the first Posiva Process Report (Rasilainen 2004); and
- the FEPs used in the recent Swedish safety assessment SR-Can (SKB 2006a).

Furthermore, in the KBS-3H safety assessment, process tables given in the KBS-3H Process Report (Gribi et al. 2007) that summarise the handling of internal processes in the safety assessment were used as check lists to ensure that no important processes and associated uncertainties had been overlooked in describing system evolution and in defining the scenarios considered in the safety assessment.

4.2 Definition and formulation of scenarios

4.2.1 Classes of scenarios

The evolution of the disposal system will be affected by FEPs internal to the system and by external FEPs, particularly climatic events and processes, all of which are subject to uncertainty. Uncertainties give rise to a range of scenarios that describe the potential future evolution of the disposal system. The classification of scenarios adopted by Posiva in the recent safety assessment is shown in Figure 4-2. This will updated for the PSAR 2012 based on guidance in YVL E.5 (see Ch. 11).

The classes of main scenarios are:

- climatic scenarios (Section 4.2.2);
- the base scenario (Section 4.2.3); and
- assessment scenarios (Section 4.2.4).

Assessment scenarios are further classified as repository assessment scenarios, dose assessment scenarios and human intrusion scenarios.

In future, the Formulation of Scenarios Report (Fig. 1-3) will describe possible paths for the evolution of the repository over time, and identify specific scenarios for detailed analysis. The Formulation of Scenarios Report is similar in scope to the earlier KBS-3V Evolution Report (Pastina & Hellä 2006) and the KBS-3H Evolution Report (Smith et al. 2007b).

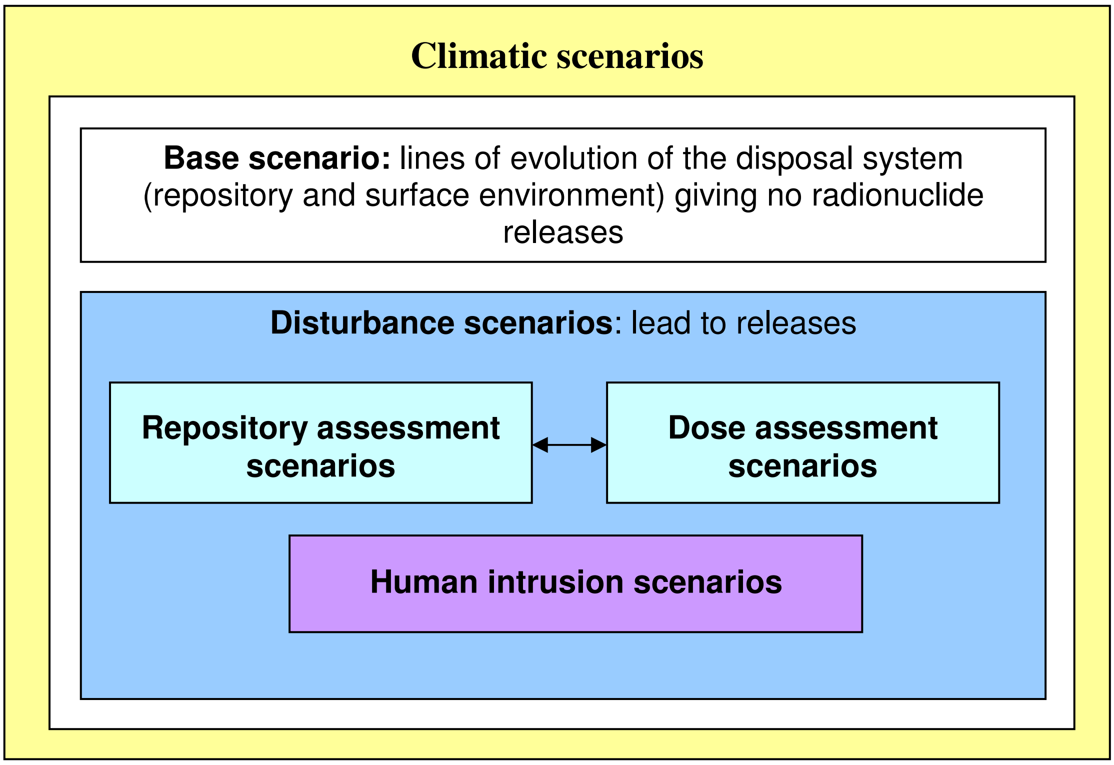

Figure 4-2. *Scenario classification in the safety assessment.*

4.2.2 Climatic scenarios

Climatic scenarios provide the framework within which the evolution of the disposal system can be described. The evolution of conditions at repository depth will be isolated from the consequences of minor climatic variations. However, major climate changes and, in particular, the formation of permafrost and ice sheets give rise to events and processes that could significantly affect the safety functions of both the engineered barriers and the bedrock. These events and processes include, for example, major post-glacial earthquakes and significant changes in groundwater composition. Climatic evolution is uncertain due, for example, to the uncertain effects of anthropogenic emissions, and a number of possible climatic scenarios must be considered to take account of such uncertainties.

Cedercreutz (2004) has identified possible scenarios for climatic evolution at Olkiluoto over the next few hundreds of thousands of years. Two climate scenarios are judged to be representative of the range of plausible possibilities for future climate development: (i), the Weichselian Repetition scenario, which is based on a repetition of conditions that are believed to have occurred during the last glacial cycle, and (ii), the Moderate Anthropogenic Emissions scenario, in which the formation of permafrost and ice sheets at Olkiluoto are delayed by climatic warming due to anthropogenic emissions, especially greenhouse gases.

According to the Weichselian Repetition scenario, the next 50 000 years will be characterised by alternating permafrost and temperate climate states, following which the area will be covered by an ice sheet, and this cover will remain for the following

25 000 years. According to the Moderate Anthropogenic Emissions scenario, a permafrost stage will be reached only after 170 000 years from now, and no ice sheet will be formed before 350 000 years. Thus, irrespective such the uncertainties associated with anthropogenic emissions, permafrost and glaciation are not expected over the period for which dose calculations are required by regulations (the first several thousands of years). However, extended periods of permafrost and glacial episodes are expected to occur in the longer term and these have to be taken into consideration in the safety case.

Climatic evolution following the development of an ice sheet is similar in both scenarios. The load of up to a 2 km-thick ice sheet is expected to depress the Earth's crust, leaving the Olkiluoto area submerged below sea level. The site will experience a gradual uplift following ice-sheet melting and return to temperate conditions, as in past glacial cycles.

Scientific understanding on climate development has increased considerably since these scenarios were formulated. Posiva has thus recently started a new project to reassess the time windows for warm and cold periods, their probabilities and possible extremes, in different climate scenarios covering the next 100 000 years, for use in future safety assessments (Posiva 2009a, Section 6.5.1).

4.2.3 The base scenario

According to Guide YVL E.5 (Paragraph A1.5), the base scenario *"… shall assume the performance targets defined for each barrier, taking account of the incidental deviations from the target values"*.

Provisional performance targets for the safety functions of the engineered barrier components and target properties for the host rock are given TKS-2009 (Posiva 2009a, Section 6.1.4). As long as the performance targets are met and target properties achieved, the canisters will remain intact and there will be no releases of radionuclides and no exposure of humans and other biota to radioactivity. Because of the favourable features of the repository described in Section 4.1.2, this is expected to be the case for the majority of canisters over a period extending into the far future, irrespective of minor deviations from design values of parameters such as buffer density and swelling pressure. In the safety analyses described in the present report, the base scenario includes all lines of evolution of the disposal system giving no release of radionuclides (see, e.g., Smith et al. 2007a, Section 8.3).

Even though the base scenario involves no radionuclide releases, the disposal system will nevertheless undergo significant changes over time in this scenario, particularly in the early, transient phase when the heat output from the fuel is high and the buffer and near-field rock are undergoing saturation. Understanding of system evolution in the base scenario is supported by process modelling or simplified scoping calculations that assess the extent to which potentially detrimental processes will affect the capacity of the system components to meet their performance targets or target properties, as described, for example, in the KBS-3H Evolution Report (Smith et al. 2007b). On the basis of these analyses, in the base scenario, the potentially detrimental features, events and processes identified in Section 4.1.3 are assumed either not to occur under the specific conditions that will prevail in the repository, or, if they do occur, to have only insignificant effects on isolation of spent fuel and radionuclide containment.

No plausible scenario has been identified whereby the changes in the surface environment affect the integrity of the repository during the first several millennia after repository closure. Thus, all lines of evolution of the surface environment during the first several millennia are included in the base scenario (this is also the minimum time window for which a quantitative dose assessment is required by Finnish regulations).

The description of system evolution in the base scenario, which is summarised in Chapter 5, includes an account of system evolution that starts at the time of emplacement of the first canister in the repository and extends to the far future.

4.2.4 Assessment scenarios

Assessment scenarios include those lines of evolution of the disposal system that include radionuclide release and hence lead to the possibility of exposure of humans and other biota to ionising radiation. They comprise repository assessment scenarios, dose assessment scenarios and human intrusion scenarios. These categories of scenarios are each described separately, below. Chapter 6 describes the current methodology to formulate assessment scenarios, and the specific scenarios analysed in the 2009 safety analysis of a KBS-3V repository and in the KBS-3H safety assessment. The analysis of these scenarios is described in Chapter 7 (KBS-3V) and Chapter 8 (KBS-3H).

Repository assessment scenarios

Repository assessment scenarios are developed for possible initial states and lines of evolution of the repository leading to radionuclide release, as a result of uncertainties in the features, events and processes listed in Section 4.1.3. These generally have a low probability of occurrence, although in some cases the probabilities are not yet well defined.

Dose assessment scenarios

Any exposure of humans and other biota to radioactivity depends on the distribution of radionuclide release locations to the surface environment, which in turn depends on how this environment evolves over time, and on future human activities that affect the surface environment. Dose assessment scenarios are scenarios describing the potential fate of radionuclides in the surface environment. They include lines of evolution of the surface environment that also form part of the base scenario, and lines of evolution for how humans and other biota inhabit and use the surface environment during the time window for quantitative dose assessment (at least several millennia), taking regulatory guidelines into account.

Human intrusion scenarios

The future occurrence of human intrusion at the site is affected by uncertainties in the evolution of human society and of the state-of-the-art in science and technology. These uncertainties are such that estimates of both probability and consequences of human intrusion scenarios must be based on “stylised assumptions” that cannot be fully substantiated or evaluated in respect to conservatism of radiological consequence estimates. Human intrusion scenarios are thus considered as a class of scenarios separate from repository assessment scenarios.

4.3 Analysis of scenarios

4.3.1 Types and uses of models and data

The evolution of the disposal system and the fate of radionuclides arising from repository assessment scenarios are analysed using models and data.

The models and data and computer codes used in the 2009 KBS-3V safety analysis and in the KBS-3H safety assessment are being documented systematically in the *Models and Data Report 2010* of the safety case portfolio (Fig. 1-3). The quality measures applied in the development and application of models, data and associated computer codes in safety assessments are also briefly described in Section 10.2.4.

The models used in safety assessment range from detailed models that aim at a realistic description of specific processes - sometimes termed “process models” to more simplified models used for analysing radionuclide release and transport in a conservative manner, i.e. one that is expected to overestimate radiological consequences. Examples of process modelling include the thermo-mechanical analyses used to investigate stress- and thermally-induced rock spalling, as well as groundwater flow modelling and modelling of thermal evolution within and around the repository.

Process models are important not only in the sense that they are part of the wider development of scientific understanding, but also because they support an evaluation of whether the repository achieves EBS performance targets and whether bedrock target properties are maintained as it evolves over time. This evaluation is part of the methodology for the formulation of repository assessment scenarios, described in Section 6.1.1. Furthermore, process modelling can support the selection of parameter values for the simplified analyses of radionuclide release and transport (e.g. the geosphere transport resistance parameter, see Fig. 4-3).

4.3.2 Calculation cases and the assessment model chain

The radionuclide releases and radiological consequences arising from assessment scenarios are evaluated by defining a range of calculation cases for each scenario, each representing different possibilities for how a system might evolve and perform over time, taking into account uncertainties in the state of the barriers, release and transport processes and in radiological exposure characteristics. In the 2009 safety analysis of a KBS-3V repository and in the KBS-3H safety assessment, as in earlier Finnish safety assessments, a purely “deterministic” approach to the analysis of scenarios has been adopted, involving:

- defining and modelling a base calculation case for the whole analysis (KBS-3V) or separate base calculation cases for each identified canister failure mode (KBS-3H);

- identifying alternative conceptual assumptions and parameter values consistent with current scientific understanding; and
- defining and modelling variant calculation cases that incorporate these alternatives either individually or in combination.

Parameters in the base calculation cases (or base cases) are, in most instances, selected to be either realistic or moderately conservative in the sense that they are expected to lead to an overestimate of radiological consequences. The variant cases for the most part take a more pessimistic view of uncertainties than the base cases. By comparing the results of a variant case analysis with those of the corresponding base case, the impact of specific model and parameter uncertainties can be illustrated. Three classes of variant repository calculation cases were distinguished in Nykyri et al. (2008):

- Sensitivity cases account for uncertainties in the knowledge of the state and behaviour of the system (or parts of it) in the long term, which is reflected in the variability of the data used in the analyses.
- "What-if" cases illustrate the impact of unlikely events and processes that could affect the repository during its evolution. The timing and magnitudes of the events and processes considered in the "what-if" cases are not necessarily regarded plausible in the present repository.
- Supplementary calculation cases aim to "check" the robustness of the system of part of the system against unexpected events and processes.

A similar classification was applied for biosphere calculation cases in the dose assessment (see Section 7.1.2 and Hjerpe et al. 2010). Calculation cases are analysed using a model chain in which, in agreement with most safety assessments carried out internationally, near-field release and transport modelling, geosphere transport modelling, biosphere transport modelling and radiological consequence analysis are considered sequentially, as illustrated in Figure 4-3.

Conservatively (in the sense defined above), it is assumed that radionuclides migrate from the repository near field (the engineered barriers and the disturbed part of the bedrock that surrounds them) to the geosphere (the remainder of the bedrock), but not vice-versa. Similarly, it is conservatively assumed that radionuclides may migrate from the geosphere to the biosphere, but again not vice-versa.

Groundwater flow modelling and surface and near-surface hydrological modelling - as described in Section 4.3.4 - supported by the descriptions of the host rock and surface environment and their evolution, provide key inputs to the principal radionuclide release and transport models shown as white boxes in Figure 4-3.

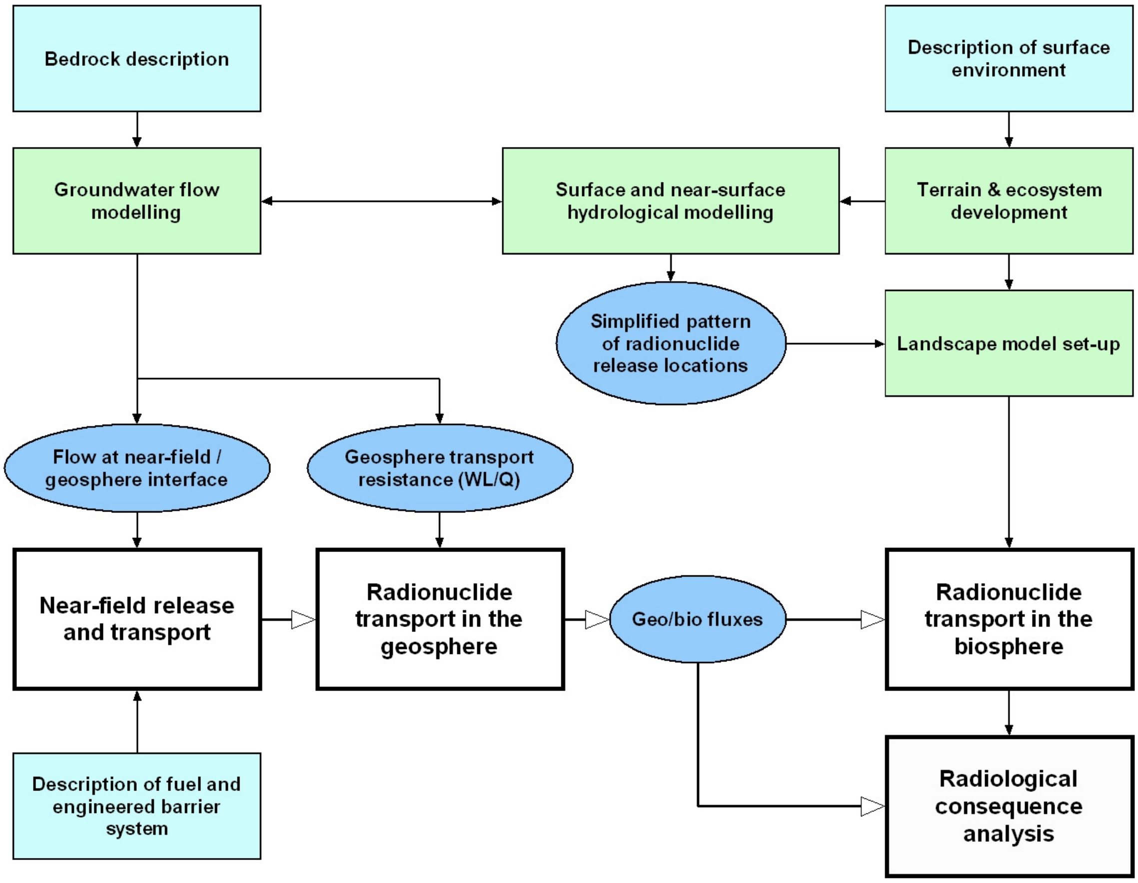

Figure 4-3. *Models and information flows. Radionuclide release and transport models and consequence analysis are shown in white boxes. System descriptions are shown in light blue boxes, key supporting models in green boxes and their principal outputs in dark blue ovals.*

Modelling of near-field radionuclide release is described in Section 4.3.5. Modelling of radionuclide transport in the near field and geosphere is described in Section 4.3.6. A systematic listing and classification of near-field and geosphere model assumptions is given in Appendix A.

Posiva's approach to assessing the radiological consequences of geosphere releases comprises the development of a description of current conditions, their evolution over time and potential radionuclide transport processes in the surface systems and, on the basis of this, (i), predicting - or forecasting - the evolution of the topography, surface hydrology, flora and fauna, by means of terrain and ecosystems development modelling (TESM) (Section 4.3.7), (ii), transport modelling using landscape models with different configurations (Section 4.3.8); and (iii), radiological consequence analysis (Section 4.4).

4.3.3 Modelling groundwater flow and surface and near-surface hydrology

Groundwater flow modelling is carried out to develop understanding of:

- undisturbed groundwater flow at the site;
- the impact of repository excavation, operation and closure on flow; and

- the long-term evolution of groundwater flow.

Simulations in support of the 2009 safety analysis of a KBS-3V repository are reported in Löfman & Poteri (2008). Updates of these simulations are reported in Site 2008 (Posiva 2009b). Modelling of groundwater flow around a KBS-3H repository at Olkiluoto is reported in Lanyon & Marschall (2006).

Groundwater flow is currently modelled using both equivalent porous medium (EPM) and discrete fracture network (DFN) approaches. Both approaches have advantages and limitations. DFN modelling includes a statistical description of the fracture distribution in the bedrock on a detailed scale. However, practical limitations of the DFN modelling carried out so far are that the density dependence of flow is not taken into account and that only steady state flow conditions have been simulated in different statistical realisations of the fracture network. EPM modelling includes deformation zones as discrete features, but otherwise applies average properties to the rock volumes between these features. It can, however, deal with transient conditions, and EPM flow simulations have thus been used to address the evolving flow conditions in the repository near field and at the site scale during repository operation and in the post-closure period.

DFN modelling is used to provide input to geosphere transport modelling, since it can be used in conjunction with particle tracking to evaluate the geosphere transport resistance, which is the main hydrogeological input parameter of the geosphere transport model. Together with surface hydrological modelling, DFN modelling also provides the release locations required for the analysis of radionuclide transport in the biosphere. EPM modelling provides input to near-field release and transport modelling. It should be noted, however, that, for steady state flow conditions, the EPM and DFN simulations show consistent results regarding the both the flow rates around the repository and the characteristics of flow and potential radionuclide transport paths at the site scale.

Simulations in support of the 2009 safety analysis of a KBS-3V repository considered flow and transport paths starting from six spatially distinct parts of the repository (panels). Although the simulations showed some differences in the distributions of geosphere transport resistances in the different panels, the distributions largely overlap, and geosphere transport resistances for the safety analysis were selected that are judged to be conservative for all panels (see Section 4.2 of Nykyri et al. 2008).

The first version of the Olkiluoto surface and near-surface hydrology model was developed during 2007 (Karvonen 2008) and further refined using data from ONKALO and from intensively monitored forested plots (Karvonen 2009a, b). In the biosphere assessment BSA-2009, the model was adapted to the future situation forecast by the terrain and ecosystems development modelling (TESM, Ikonen et al. 2010), and used to continue the release paths from the DFN simulations through the overburden into the rooting zone or surface water bodies (Karvonen 2009c). The model also provides the landscape model with the water balance data for each discrete part of the modelled area (“biosphere object”).

A summary of results of groundwater flow modelling and surface and near-surface hydrological modelling is presented in Section 5.1.

4.3.4 Modelling radionuclide release from the fuel

Following canister failure and contact of the fuel with water, there will be a relatively rapid release to solution of the radionuclide inventory at grain boundaries and in gaps. This part of the radionuclide inventory is termed the instant release fraction, or IRF, since its release to solution is conservatively modelled as instantaneous. Release of the IRF will be followed by a slower, congruent release of radionuclides from the fuel matrix, the cladding and other metal components.

It is assumed in safety assessment modelling that the inventory of activation products in Zircaloy and other metal parts is released congruently with the corrosion of the metal. The fractional corrosion rate of Zircaloy is taken to be 10^{-4} per year, with 10^{-3} per year for other metal parts. These values are taken from TILA-99 (p. 101 of Vieno & Nordman 1999).

The slow, long-term release of radionuclides from the fuel matrix is handled in safety assessment modelling by assuming a constant fuel degradation rate, leading to constant release rates of radionuclides from the matrix. Werme et al. (2004) reviewed the available data for relevant reducing conditions inside a canister and proposed a fractional degradation rate of 10^{-6} to 10^{-8} per year. As noted in Section 4.1.2, this range is consistent with recent experimental results of Grambow et al. (2008). For the purpose of the 2009 safety analysis of a KBS-3V repository and for the KBS-3H safety assessment, a central value of 10^{-7} a^{-1} was used in most of the calculation cases, except for a few alternative calculation cases, where 10^{-6} a^{-1} was applied.

Radionuclides will either enter solution, form volatile species that can mix with repository-generated gas (relevant for C-14), or, if their solubilities in water are low, precipitate either as immobile solids or as colloids. They may also sorb onto solid surfaces, such as those of the iron oxyhydroxides formed by corrosion of the iron insert of the canister, but this possibility is conservatively omitted in modelling.

The solubility limits used in the 2009 safety analysis of a KBS-3V repository and in the KBS-3H safety assessment are reported in Grivé et al. (2007). Several elements (Ni, Se, Sr, Zr, Nb, Pd, Sn, and Mo) can be found as stable isotopes forming part of e.g. the fuel rods and bundles, or can be generated by decay of some radionuclides. The amounts of these stable isotopes can determine whether or not the solubility limits are reached, as they tend to be larger than the amounts of radioactive isotopes of the same elements. This “sharing” of solubility limits is included in near-field modelling of radionuclide release.

4.3.5 Modelling radionuclide transport in the near-field and geosphere

Dissolved radionuclides and radionuclides in gaseous form are modelled as uniformly distributed in the void space in the canister interior, from where they migrate into the surrounding buffer, and from there into the geosphere. If the canister fails due to the presence of a small hole (e.g. a penetrating defect in the weld), the transport resistance provided by this hole can be taken into account in modelling release rates from the canister. The most important radionuclide transport and retention processes, which are included in near-field and geosphere transport models, are advection, diffusion and sorption, as described in the following paragraphs.

Advection and diffusion

If the buffer performs in accordance with its performance targets, the dominant radionuclide transport process occurring within the buffer is diffusion. If the gas pressure inside the canister exceeds the gas breakthrough pressure of bentonite, advective gas pathways will also form that may transport any C-14 present as volatile species. These will reseal when the gas pressure drops, and are not expected to affect the transport of dissolved radionuclides significantly. Radionuclides may be present inside the canister in colloidal form, but these will be prevented from migrating through the buffer by its microporous structure (see, e.g. Section 3.3.5 of Miller & Marcos 2007).

Diffusion equations describing dissolved radionuclide migration in the buffer are given Liu & Neretnieks (1997) and in SKB (2006b). The process has been thoroughly studied. Liu & Neretnieks (1997) give a detailed discussion of different experimental methods to quantify diffusion and how the results can be interpreted. The diffusivity data for radionuclides through compacted bentonite compiled for SR-Can (SKB 2006c) have also been used in 2009 safety analysis of a KBS-3V repository and in the KBS-3H safety assessment.

In the KBS-3V safety analysis, radionuclides were modelled as being transferred from the buffer to the geosphere by three alternative paths, as illustrated in Figure 4-4 (in the figure, canister failure is indicated as occurring near the top of the canister, which could be due, for example, to the presence of a penetrating defect in the canister weld):

- from the buffer directly to a host rock fracture intersecting a deposition hole (Q_F);
- from the buffer to the backfill in the upper part of the deposition hole and thence to the deposition tunnel EDZ (Q_{DZ}); and
- from the buffer to the deposition tunnel backfill and thence to its EDZ (Q_{TDZ}).

In the KBS-3H safety assessment, only a single path was considered - from the buffer directly to a host rock fracture intersecting the deposition drift near to the canister failure location – as illustrated in Figure 4-5. Again, the rate of transfer to the geosphere is controlled by an effective flow rate.

In the repository calculation cases in the KBS-3V safety analysis and in the majority of cases in the KBS-3H safety assessment, phenomena that could potentially perturb mass transfer at the buffer rock interface, such as the presence of the KBS-3H supercontainer shell and its corrosion products, have been assumed to have a negligible effect, although a few calculation cases have been considered where the mass transfer properties of the interface have been treated more pessimistically (Section 8.2.2).

The rate of transfer to the geosphere is controlled in each case by effective flow rates, Q_F, Q_{DZ} and Q_{TDZ}, respectively, determined from the results of EPM modelling (see Section 4.3.4).

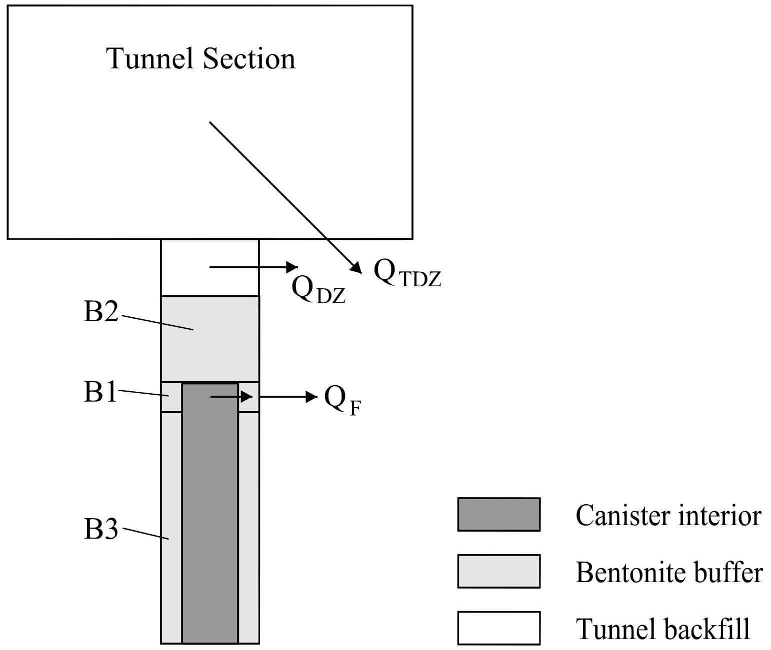

Figure 4-4. *Conceptual near-field transport model for a KBS-3V repository (after Fig. 11-3 Vieno & Nordman 1999).*

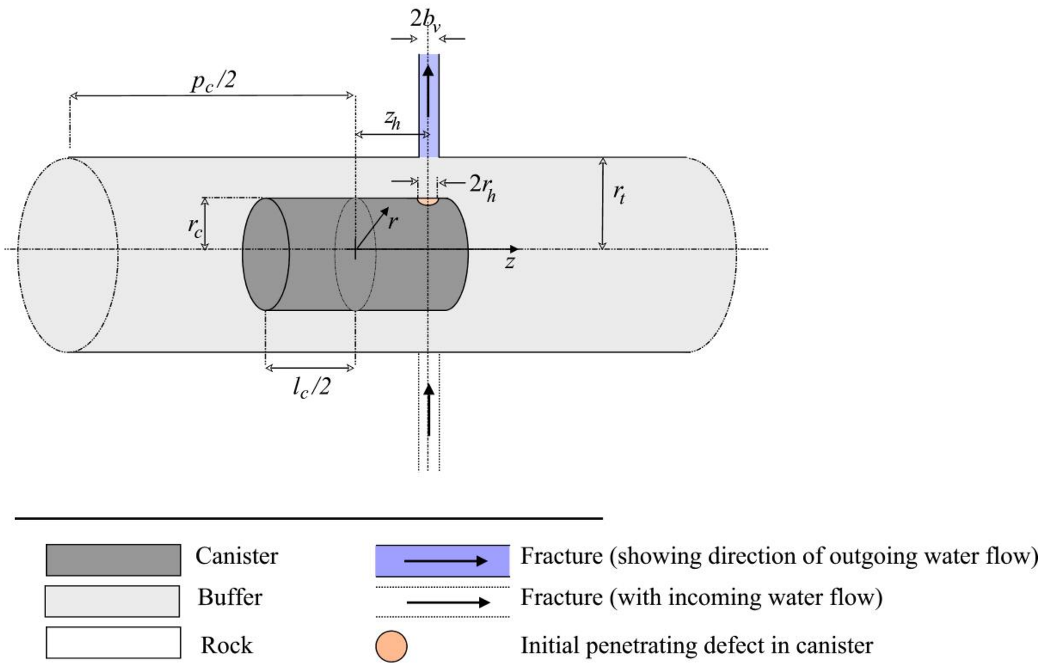

Figure 4-5. *Conceptual near-field transport model for a KBS-3H repository. The values of the geometrical parameters shown in the figure are given in the KBS-3H Radionuclide Transport Report (Smith et al. 2007c).*

The conceptual geosphere transport model is sown in Figure 4-6. Radionuclides that enter the geosphere in either dissolved or gaseous form are modelled as being transported through transmissive fractures predominantly by advection, i.e. by the bulk motion of flowing groundwater. The radionuclides migrate along transport paths in the bedrock each characterised by transport parameters that are assumed identical for modelling purposes. Variations in the flow field within individual fractures coupled with mechanical mixing and molecular diffusion can result in local water flows and velocities that differ from the larger-scale average, resulting in the spreading of radionuclide releases in a process termed hydrodynamic dispersion. Hydrodynamic dispersion has not been included in safety assessment modelling for the 2009 safety analysis of a KBS-3V repository or for the KBS-3H safety assessment. However, inclusion of this process was found to have only a minor effect on radiological consequences in TILA-99 (Section 12.8 in Vieno & Nordman 1999).

In fracture planes and in the wall rock adjacent to them, there are numerous water-filled void spaces and rock matrix pores where water is effectively stagnant. Here, diffusion of solutes (matrix diffusion) is the dominant transport mechanism (Neretnieks 1980). Matrix diffusion coupled to sorption on matrix pore surfaces is an important retention mechanism in geosphere transport modelling for many radionuclides. Even for non-sorbing species, matrix diffusion may provide an efficient temporal retardation and spreading process simply by removing these species from the flowing groundwater.

Sorption

Sorption is a general term describing the attachment of dissolved species to mineral surfaces. It includes ion exchange, physical adsorption and surface complexation.

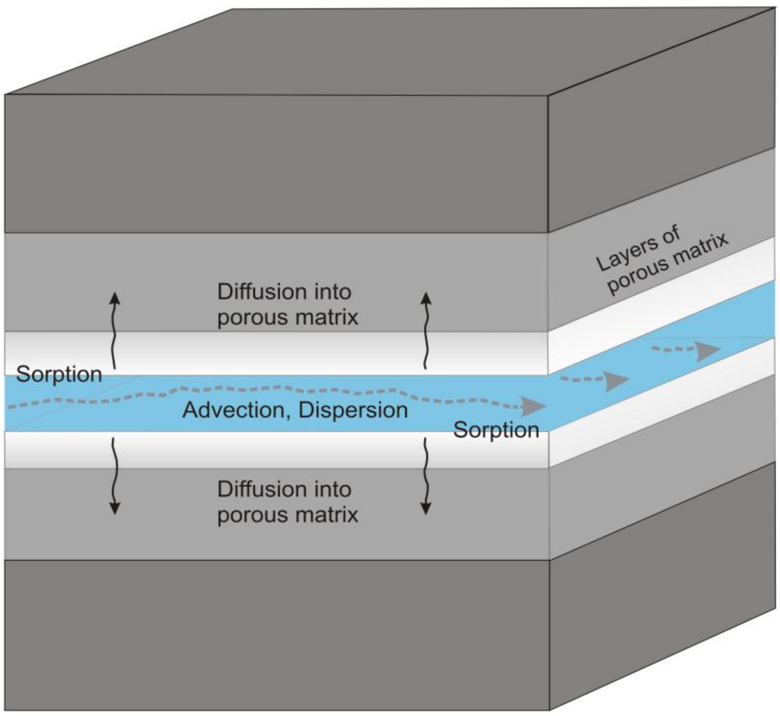

Figure 4-6. *Conceptual geosphere transport model (Fig. 5-11 in Posiva 2010).*

Sorption is element specific and depends on both the aqueous speciation of radionuclides, and the solid phase composition and surface characteristics.

In transport modelling, the sorption of radionuclides on the pore surfaces of the buffer and rock are taken into account. Equilibrium sorption is assumed (i.e. kinetics are fast compared with transport processes), such that the degree of sorption may be quantified by a species-dependent distribution coefficient (K_d). K_d is strongly dependent on the prevailing chemical conditions. Its value may change, e.g. if the redox conditions or temperature changes, if the sorbing solid surfaces dissolve or if they undergo other mineralogical transformations.

K_d values of different radionuclides in bentonite have been thoroughly studied by many different laboratories and different databases have been compiled by the various waste management agencies (e.g. Bradbury & Baeyens 2003 for Nagra and Ochs & Talerico 2004 for SKB).

4.3.6 Forecasting the evolution of surface environment

Forecasts of the evolution of the conditions of surface systems at Olkiluoto are based on the results of terrain and ecosystems development modelling (TESM) and surface and near-surface hydrology modelling (Section 4.3.4). The terrain and ecosystem development model report (TESM-2009; Ikonen et al. 2010) provides an up-to-date scientific synthesis of the expected evolution of the surface systems for the time period when the dose-based constraints apply. TESM-2009 is based on the latest available site-specific data and models, such as the terrain (topographical) model (Pohjola et al. 2009), the land uplift model (Påsse 2001, revised by Vuorela et al. 2009). It is an update of TESM-2006 (Ikonen 2007), and will be further updated in 2011.

The results of TESM are used to define the initial state of the biosphere and its subsequent evolution (Section 5.1). They also provide the starting point for the modelling of radionuclide transport in the biosphere (Section 4.3.7).

In TESM, lakes, rivers and their catchment areas are identified using standard GIS (geographical information system) processing tools, applying an approach found suitable for the site by Ojala et al. (2006). Evolving terrestrial and aquatic erosion and sedimentation are accounted for in the predictions, as is accumulation of organic material in reed beds and wetlands. The evolving thickness of the humus layer is predicted by the vegetation modules of the model. The assumed habitats of characteristic groups of animals are based on identification of different ecosystems in the evolving landscape (Chapters 4-9 in Haapanen et al. 2009a).

Future human settlement and land use are highly uncertain, and predictions of these made by TESM are intended only as illustrations of potential evolutions. Illustrative simulations of human settlement are based on correlations with various factors affecting the housing density around the Olkiluoto site (e.g. soil type and distances to main roads, nearest neighbour or a water body). The future locations of croplands are based on soil suitability and the current agricultural preferences in the region (Ikonen 2007).

4.3.7 Modelling radionuclide transport in the biosphere

Biosphere objects and landscape modelling

Radionuclide transport in the biosphere is modelled so as to obtain radionuclide-specific spatial activity distributions in the various ecosystems that may be present at the site. The ecosystems that may receive radionuclides from the geosphere are first identified, based on the terrain and ecosystems development modelling (TESM) and on radionuclide release patterns derived from groundwater flow modelling and surface and near-surface hydrology modelling. For transport modelling purposes, the ecosystems are modelled as discrete units, termed *biosphere objects*, characterised by the ecosystem type, i.e. *forest*, *wetland*, *cropland*, *lake*, *river* and *coast*, and an associated set of parameter values. Biosphere objects are connected to each other, in the sense that terrestrial materials and water, together with their associated radionuclides, can migrate from one to another. The nature of the connections is also inferred from the results of TESM and the surface and near-surface hydrological modelling. The connected set of biosphere objects is termed the *landscape model*, which may be configured in different ways to account for uncertainty and variability with time.

Biosphere modelling in the 2009 safety analysis of a KBS-3V repository

In the 2009 KBS-3V safety analysis, a three-tiered graded approach was used. The graded approach consists of using the results at lower tiers (1 and 2) to identify radionuclides that are highly confidently expected to have insignificant radiological consequences, and thus do not need to be considered at the highest tier (Tier 3).

The quantity of interest in Tiers 1 and 2 is the Risk Quotient (RQ), which is the calculated nuclide-specific dose rate, divided by pre-selected screening dose rates (SDR). The approach must be sufficiently cautious that there is a high degree of confidence that the potential radiological consequences are below the relevant regulatory requirements when the RQ is below 1. The SDRs are selected to 10 nSv per year for humans, which is two orders of magnitudes below the lowest regulatory dose constraint, and 0.01 mGy per hour for the other biota, which is the default generic screening absorbed dose rate in the ERICA Tier 1 (see below) and recommended by the PROTECT project (Andersson et al. 2008). These are considered low enough to allow Tier 1 and 2 to screen out individual radionuclides for which the RQ is below 1.

In Tier 1, an extremely cautious approach is taken, in which it is assumed that a hypothetical individual receives the maximum exposure over one year to the whole integrated release from the geosphere. In Tier 2, a screening model is applied, which includes a higher degree of realism than the model used in Tier 1, but is still sufficiently cautious for the screening purpose. The generic model includes two generic ecosystem-specific sub-models, one terrestrial and one aquatic, and a well sub-model. The screening model used in Tier 2 does not require site-specific parameters and therefore can be called “generic”. If the RQ calculated for a specific radionuclide in Tier 2 is greater than 1, then it is necessary to consider that radionuclide in the site-specific landscape modelling (Tier 3).

The screening evaluation, using increasingly complex and realistic models in successive tiers, avoids the need to determine numerous site-specific parameters for radionuclides that are in any case unimportant.

The graded approach facilitates and strengthens the confidence in the biosphere assessment and strengthens the demonstration of compliance with regulatory criteria, especially by increasing the transparency of the biosphere assessment. Furthermore, it is resource saving and also provides an instrument for analysing model uncertainties, and providing guidance for the development of the landscape model, and the associated environmental monitoring programme. Some results of this tiered approach are given in Section 7.4.

Biosphere modelling in the 2009 safety assessment of a KBS-3H repository

The biosphere assessment for the KBS-3H safety assessment was carried out before the above-mentioned screening procedure had been developed. The landscape model was thus used to evaluate the annual landscape dose (annual effective dose), and typical absorbed dose rates to other biota, due to all radionuclides released according to the geosphere assessment model, for all relevant repository base and sensitivity calculation cases (Section 8.5).

4.3.8 Treatment of uncertainty

Uncertainty arises from imperfect knowledge of the system to be assessed and its evolution, and is inherent in all safety assessments and safety cases. An integral part of a safety case is an examination of how strongly the parameter values, conceptualisations and theoretical assumptions used by models in various parts of the safety case affect model outcomes, and to identify the sources of uncertainty that have the biggest impact on the outcome of the analysis.

Uncertainties that could lead to failure to achieve performance targets or target properties and hence, potentially, to loss of, or significant perturbation to, the safety functions have been mentioned in the context of the formulation of assessment scenarios, above. Scoping calculations play an important role in examining the impact of specific uncertain processes or events on these performance targets and target properties. Examples are given in Chapter 6 (Section 6.2.1 and 6.2.2).

As described above, the radionuclide releases and radiological consequences arising from assessment scenarios are evaluated by defining a range of calculation cases for each scenario, each representing different possibilities for how a system might evolve and perform over time. For any given calculation case, the uncertainty in the results of radionuclide transport modelling can be divided into two categories: i) those caused by uncertainty in the input data and ii) those caused by uncertainty in the modelling process itself. Uncertainties in input data are generally treated by appropriate parameter value selection. The various assumptions underlying the near-field and geosphere modelling processes are summarised in tables in Appendix A.

In individual calculation cases, specific model assumptions may and parameter values are selected based on current scientific understanding of FEPs and their uncertainties, and may be realistic, pessimistic or conservative according to this understanding. Where multiple parameter values, or alternative model assumptions, are considered in the calculation cases, a comparison of results illustrates the impact on the corresponding uncertainties on calculated radiological consequences.

4.4 Radiological consequence analysis and safety indicators

Different regulatory requirements on the radiological protection are given in the Government Decree on the safety of disposal of nuclear waste (DG 736/2008) and in the Guide YVL E.5 for different time windows. In the earliest post-closure time window, for a period that shall extend at a minimum over several millennia, Section 4 of DG 736/2008 states:

"The annual dose to the most exposed people shall remain below the value of 0.1 mSv.

and

"The average annual doses to people shall remain insignificantly low".

Guide YVL E.5 states in Para 3.10 that:

"The annual dose for the most exposed individuals, 0.1 mSv per year, stands for an average dose e.g. in a self-sustaining family or small village community living in the environs of the disposal site, where the highest radiation exposure arises via the various pathways".

and in Para. 3.11:

"In addition, the average annual doses to such larger groups of people shall be addressed, who live at a regional lake or a coastal site and are exposed to the radioactive substances transported via these watercourses. The acceptability of these doses depends on the number of exposed people, however, these doses shall not be more than one hundredth to one tenth of the constraint for the most exposed individuals".

It is also stated (Guide YVL E.5 Para. 3.18) that:

"Disposal shall not affect detrimentally to species of fauna and flora".

In the current Posiva methodology, activity concentrations derived from landscape modelling are used for assessing potential radiological consequences to humans and other biota. The assessment of dose is made based on the most recent international recommendations (ICRP 2000, 2007). Four exposure pathways are considered (ingestion of food, ingestion of water, inhalation and external exposure). The properties (size and location) of the exposed population are determined by the capability of producing food and drinking water at the site, size of suitable residential areas from a present-day perspective, and present demography.

In the longer term, the quantitative regulatory criteria relate directly to the release rates of radionuclides from the geosphere to the biosphere ("geo-bio fluxes"). According to Guide YVL E.5, Paragraph 3.14:

"*The sum of the ratios between the nuclide specific activity releases and the respective constraints shall be less than one*".

According to the same paragraph of YVL E.5, activity release rates may be averaged over 1 000 years at most. The regulatory constraints with which the activity release rates are compared are specified in Guide YVL E.5 Paragraph 3.13.

Geo-bio fluxes are provided directly by the results of near-field and geosphere assessment modelling. Near-field and geosphere assessment models are applied over a period extending to a million year in the future. The Olkiluoto site is judged to be stable and broadly predictable over this time frame. Furthermore, according to the NEA's Integration Group for the Safety Case, a million years seems to be emerging as a commonly accepted time frame in recent safety assessments (NEA 2007). A time frame of one million years for quantitative safety analyses is also consistent with Finnish regulations. According to Guide YVL E.5 (Para. A1.9):

"*… safety evaluations extending beyond a time horizon of once million years can be based mainly on the complementary considerations.*"

where these may include:

"*… e.g. analyses by simplified methods, comparisons with natural analogues or observations of the geological history of the disposal site*".

Geo-bio fluxes, annual effective doses from landscape modelling, and typical absorbed dose rates to assessment species[8] are the main quantities used to assess compliance with regulatory protection criteria. Other quantities are, however, also calculated with the purpose of building understanding of, and confidence in, the outcome of the safety analyses. The various calculated quantities are called safety indicators and complementary safety indicators. Safety indicators are quantities that can be directly compared with regulatory criteria and have the main role of building confidence in the compliance assessment based on landscape modelling, through the use of relatively simple exposure scenarios. The main role of the complementary safety indicators is to develop understanding of the behaviour of the biosphere.

In the KBS-3H safety assessment, two safety indicators were considered in addition to geo-bio fluxes and the annual effective dose rates derived from landscape modelling (Broed et al. 2007). These safety indicators were annual effective doses calculated using two indicative stylised well scenarios: one drinking water well (WELL-2007) and one agricultural well, where the resulting dose also include contributions from consumption of contaminated crops and animal products due to using the well water for irrigation of

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[8] A group of species selected to cover different roles in the ecosystem, assumed to cover also the other species in the trophic compartment to a reasonable degree.

crops and watering cattle, (AgriWELL-2007). The drinking water well scenario was updated for the safety analysis of a KBS-3V repository (Nykyri et al. 2008, Hjerpe et al. 2010). The agricultural well scenario was also updated (Hjerpe et al. 2010). Calculation of these doses further facilitates comparison with regulatory dose criteria, without the need for detailed biosphere modelling. Given the stylised nature of the well scenarios, however, comparison of these quantities with regulatory constraints is not regarded by itself as an adequate test of compliance with regulations.

Several complementary safety indicators have also been calculated in the both the KBS-3H safety assessment and the 2009 safety assessment of a KBS-3V repository, such as activity concentrations in soils, water, and sediments.

4.5 Assessment codes

In the 2009 safety analysis of a KBS-3V repository and in the KBS-3H safety assessment, modelling of near-field release and transport was performed with the REPCOM code. REPCOM has been developed by the Technical Research Centre of Finland (VTT) for radionuclide transport analyses in the near field of repositories for low- and intermediate-level waste and spent fuel. The REPCOM code is described in Appendix 2 of Nykyri et al. (2008) and in Appendix A of the KBS-3H Radionuclide Transport Report (Smith et al. 2007c).

Modelling of geosphere transport was performed with the FTRANS code (FTRANS 1983; Nordman & Vieno 1994). The FTRANS code is also described further in Appendix 2 of Nykyri et al. (2008) and in Appendix A of Smith et al. (2007c). Posiva is currently exploring the use of the GoldSim probabilistic simulation software for integrated near-field and geosphere transport modelling, including probabilistic sampling of parameters from PDFs[9].

The modelling tools used in biosphere assessment are summarised in BSA-2009 (Hjerpe et al. 2010) and supporting reports (Hjerpe & Broed 2010, Karvonen 2008): A toolbox called UNTAMO, run as an overlay on the ArcView software, is used for the terrain and ecosystem forecasts (Ikonen et al. 2010) and Pandora, a simulation tool based on the Matlab/Simulink environment, is used for simulating radionuclide transport in the biosphere (Åstrand et al. 2005). Auxiliary post-processing scripts are used for the dose assessment for humans and other biota (Hjerpe & Broed 2010).

Quality assurance procedures applied to these assessment codes are described in Section 10.2.4.

[9] www.goldsim.com.

5 DISPOSAL SYSTEM EVOLUTION IN THE BASE SCENARIO

This chapter (Fig. 5-1) presents a description of the base scenario for the evolution of a KBS-3V or KBS-3H repository at the Olkiluoto site.

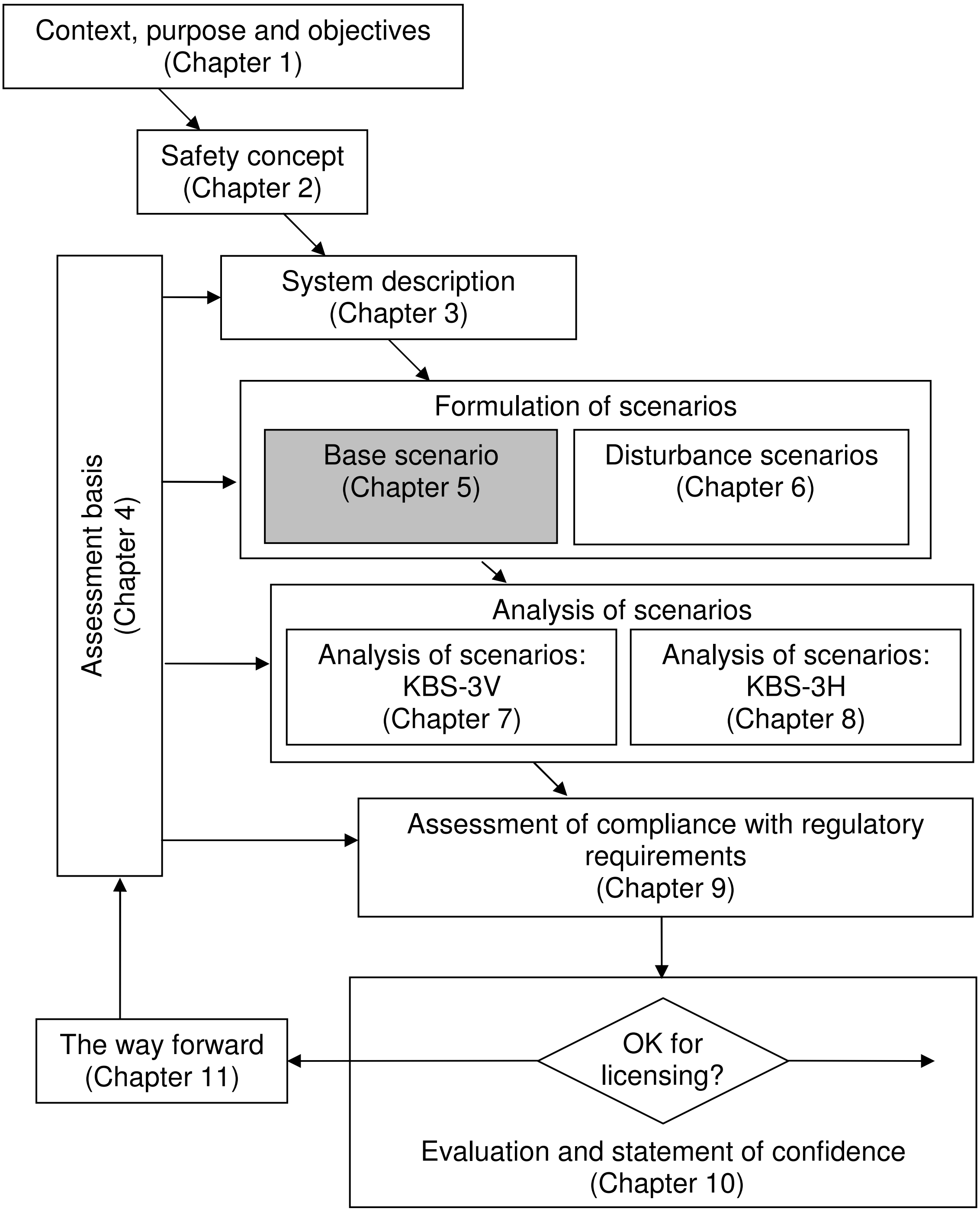


Figure 5-1. *The present chapter in the context of the safety case summary report.*

The chapter is structured as follows:

- Section 5.1 describes the initial state and early evolution of the disposal system in the transient period when the system is influenced by disturbances created by repository construction and emplacement of the spent fuel;
- Section 5.2 deals with evolution during the following period when a temperate climate is expected to prevail; and
- Section 5.3 deals with the timing and impact of major climate changes on repository evolution.

Disposal system evolution in the base scenario is described in detail in the KBS-3V and KBS-3H Evolution Reports (Pastina & Hellä 2006 and Smith et al. 2007b) and in the terrain and ecosystems development reports (Ikonen 2007, Ikonen et al. 2010). In future, potential paths for system evolution will be described in the Formulation of Scenarios Report (Fig. 1-3).

5.1 The initial state

The starting point for the description of disposal system evolution in the base scenario, and also in other scenarios, is a description of its initial state. The conditions for the surface environment predicted for the year 2020, upon the emplacement of the first canister, define the initial state of the biosphere, and are based on forecasts from the terrain and ecosystems development modelling. The initial state of the repository differs somewhat compared with the surface environment, in that, in the case of the repository near field and far field, it is specific to each canister and its position. In particular, the initial state for an individual canister and its position relates to the conditions prevailing when the buffer and, in the case of the KBS-3V variant, the backfill are installed[10]. Because the repository is implemented over a period of about 100 years, the near field and far field in different parts of the repository will be in their initial states at different times. The far field comprises the host rock. The initial state for the host rock can be divided in two phases; first the state of the host rock at the time of emplacement of the first canister has to be defined and, as a second phase, the expected range of conditions at single canister positions is determined taking into account the stage of repository construction, operation and heat production of the already emplaced canisters.

Safety related design requirements (including rock suitability criteria - RSC) constrain the range of possibilities for the initial state. They are developed with a view to ensuring that, as far as possible, performance targets are met both initially and as the system evolves. The initial state descriptions of the disposal system, as given by the production lines description in (Posiva 2009a, Chapter 5), serve, therefore, as a starting point for the description of the evolution scenarios of the disposal system and thus for long-term safety assessment.

The initial state of the host rock at the time of emplacement of the first canister takes account of the effects of the disturbances caused by ONKALO and repository

[10] This is also the point when the direct influence of repository operation procedures on the disposed canister and buffer, and also the likely possibilities for monitoring the state of the canister and buffer, have come to an end (Posiva 2009a).

construction. These disturbances include a transient drawdown of the groundwater table and an associated reduction of the hydrostatic groundwater pressure at repository depth; the upconing of saline and methane-rich water from deep underground is also possible at this stage. However, the magnitude of these hydrogeological and hydrogeochemical disturbances will be limited by the grouting of the more significant transmissive fractures intersecting underground openings. The stress changes in the rock caused by repository excavation will lead to some irreversible structural effects, including the formation of excavation damaged zones (EDZs) and also, potentially, mechanically induced rock spalling. After emplacement, the heat generated by spent fuel will cause the rock to expand, which may result in thermally-induced rock spalling (Lönnqvist & Hökmark 2007, Hakala et al. (2008). EDZ properties and the properties of rock zones where spalling has occurred are subject to considerable uncertainties and will require further investigation in future project stages. Plans are presented in TKS-2009 (Posiva 2009a, Section 6.5.7.2). These mechanical effects may change the hydraulic and transport properties of the rock surrounding the deposition holes. These changes are taken into account in groundwater flow modelling, including modelling of the transport of dissolved chemical species. Groundwater flow and salinity transport simulations have been carried out in support of the safety analysis of a KBS-3V repository (Nykyri et al. 2008, Löfman & Poteri 2008), based on the 2006 site description of Olkiluoto (Andersson et al. 2007). Updates of those simulations are reported in Site 2008 (Posiva 2009b).

5.2 Early evolution

The repository will evolve from its initial state through an early, transient phase towards a target state, in which key safety relevant physical and chemical characteristics (e.g. temperature, buffer density and swelling pressure) are subject to much slower changes than in the transient early phase. The main transient processes occurring within and around a deposition hole in a KBS-3V repository are illustrated in Figure 5-2 (see Chapters 6 and 7 in Pastina & Hellä 2006). Largely similar processes occur within a KBS-3H deposition drift (see Chapter 5 in Smith et al. 2007b), although there are some processes that are specific to each variant, and it is in the early, transient phase that most of the significant differences in evolution between the KBS-3V and KBS-3H variants arise. For example, particularly in tight KBS-3H drift sections, the gas generated by the corrosion of steel components external to the canister (principally the supercontainer shell in the current reference design) may accumulate at the buffer / rock interface, resulting in a prolonged period during which inflow of water from the surrounding rock will be limited, which will delay saturation of the buffer.

Thermal evolution in response to the heat generated by spent fuel has been modelled for both the KBS-3V (Ikonen 2003a, 2005b) and KBS-3H (Ikonen 2003b) variants. The heat generated by spent fuel will affect the temperature in the near field of the repository as indicated in Figure 5-3. The maximum temperature in fuel, canister and the near field is reached in 10-15 years after canister deposition. The maximum temperature in the rock in the vicinity of a single canister is reached in a few decades. Its magnitude depends on fuel type, repository layout - notably the density of canister emplacement - and the local thermal conductivity of the rock. As noted in Section 3.4.1, the repository layout will be adjusted to ensure acceptable repository temperatures.

The thermo-hydro-mechanical and chemical (including microbiological) processes will be interconnected (coupled) during evolution. For example, the increase of temperature in the near field will also affect groundwater flow to some extent and, as mentioned above, may lead to thermally-induced rock spalling. Groundwater flow modelling (EPM simulations) indicate that the temperature rise due to heat generation by the fuel, which decreases the viscosity and density of water, results in a buoyancy effect that has an impact on the magnitude and direction of the flow rates around the repository. The high temperature gradients tend to change the downward, topographically driven flow to either an upward flow or flow along the deposition tunnels, depending on the hydraulic conductivity of the tunnel backfill and surrounding damaged zone (damage may be caused by excavation - the EDZ - or by heat from the waste - thermally induced rock spalling), relative to that of the host rock.

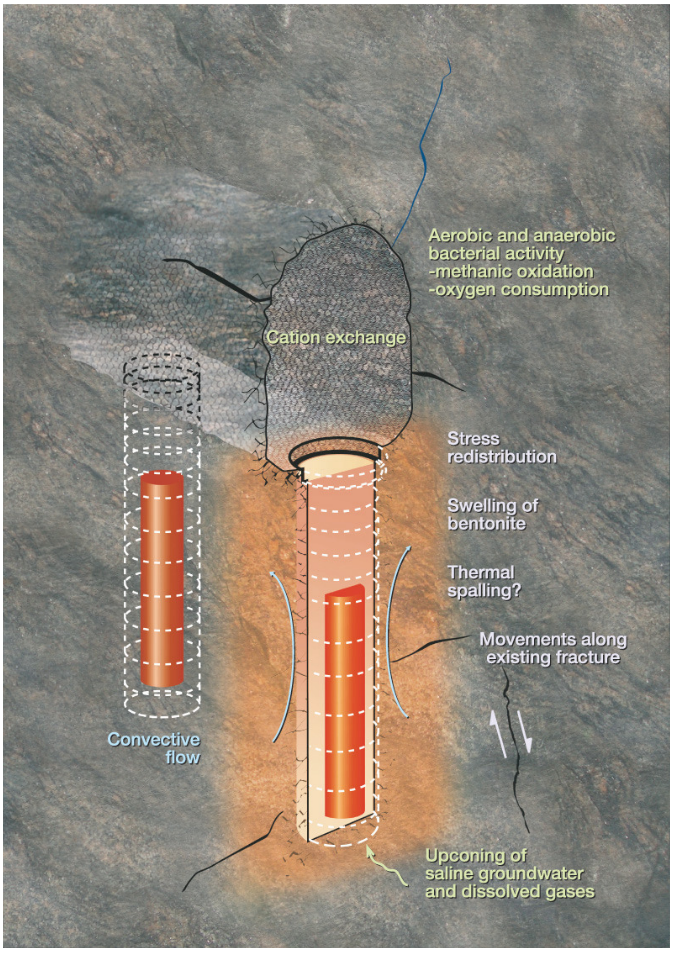


Figure 5-2. *Main processes in the deposition hole during the early evolution of a KBS-3V repository. Geologic features (e.g. rock stresses and fractures) are not to scale and are exaggerated for clarity (after Fig.7-7 in Pastina & Hellä 2006).*

The saturation and swelling of the buffer will also be affected by the heat from the canister and the availability and quality (salinity, high pH cement leachates) of water coming from water-conducting fractures. The saturation of the buffer in the KBS-3H concept will be also influenced by the gas generated by the corrosion of the supercontainer shell, which is expected to continue over a period of a few thousand years (Gribi et al. 2007). At the same time the extent of thermal spalling in the rock will be also affected by the rise of temperature and the capability of the backfill (in the KBS-3V) and the buffer to swell and form a tight contact with the rock. Silica in the buffer close to the canister is expected to dissolve during the period of elevated temperature and be transported outwards by diffusion to colder parts where precipitation may take place. Buffer cementation could in principle take place due to the dissolution, transport and precipitation of silica or aluminosilicate minerals, but neither experimental nor natural analogue studies have shown that this process will occur to any significant extent. The effect of buffer cementation due to silica precipitation is, however, an issue for further work. Saturation of the buffer and KBS-3V backfill has been modelled by Hökmark (2004), Börgesson & Hernelind (1999, 2006b), Börgesson et al. (2006) and Lempinen (2006a, b, c, d). Saturation of the buffer in KBS-3H has been modelled by Börgesson et al. (2005).

As the buffer saturates and swells, it will exert increasing and uneven mechanical loads on the canisters. The impact of these uneven loads on the canister and on the canister welds has been demonstrated to be negligible (see Section 4.2.3 of Miller & Marcos 2007). Any differences in buffer density and swelling pressure around individual canisters and also along the drift and deposition tunnels are expected to diminish over

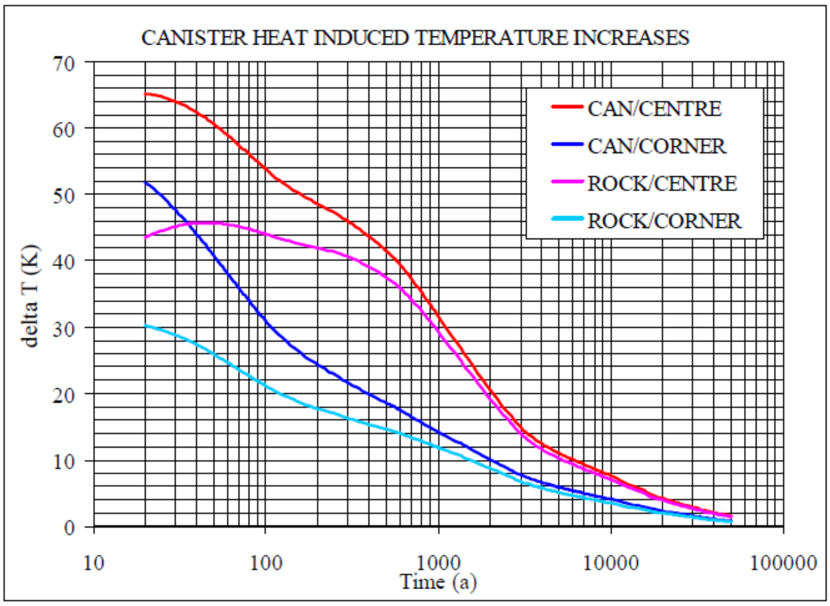


Figure 5-3. *Canister and bentonite buffer temperature increase above rock ambient (+10.5 ºC) in KBS-3V repository after closure. Can/centre and can/corner indicate a canister near the centre or near the corner of the repository, respectively. Rock/centre and rock/corner indicate rock temperatures near the centre or the edge of the repository, respectively (after Fig. 6-1 of Pastina & Hellä 2006).*

time due to homogenisation of the bentonite and, although some heterogeneity may remain, the load on the canisters is expected to become approximately isostatic (i.e. equally large over the entire canister surface area) and similar for all canisters.

In the current base scenario, the copper coverage of the canisters remains intact during the period of early evolution and beyond. As a consequence, the radionuclides associated with spent fuel are totally contained within the canisters in this scenario. It should be noted that quality control procedures during encapsulation and emplacement will aim to reduce the probability of occurrence of initial penetrating defects or defects that could lead to subsequent failure. The research, technical design and development work on canister welding and on inspection techniques for the canister welds aims at defining the upper limit for the probability of occurrence of such defects by 2011.

During the period of early evolution, the coastline will be displaced away from the Olkiluoto site (Fig. 5-4) due to continuing land uplift, even if global average sea level rises as a result of climate warming (though in this case the apparent land uplift will be slower than indicated in Fig. 5-4). As a result, the surface environment will change and the increased fresh water infiltration, together with the corresponding decreased contribution from seawater, will change the groundwater composition. No great change in the biota present at the site is expected, although anthropogenic warming may introduce some new species from further south, and also lead to an increase in the length of the growth season. In the longer term, peatlands will become abundant both by primary mire formation and by overgrowth of shallow water bodies, unless future land use disturbs the natural vegetation succession.

5.3 Evolution during the temperate climate phase

Following the period of early evolution, the repository and its geological environment will evolve to a quasi-steady target state, in which the engineered barrier performance targets and the host rock target properties are met and in which key safety relevant physical and chemical characteristics are subject to relatively slow evolution.

By definition, the repository will be saturated and heat output of fuel will have declined to a level that has no significant effect on the evolution of the repository. Minor variations in climate prior to any future glacial episode are also not expected to have any significant impact on temperatures at repository depth. The reduction in thermal output from the fuel will result in rock stress levels returning to those prior to spent fuel emplacement and in a reduction in the thermo-mechanical loads on the canister.

The buffer will continue to evolve chemically. In the case of a KBS-3H repository, mineral transformation of the buffer due to the presence of Fe(II) from corroding steel components is also likely to extend further into the buffer, but, according to the preliminary modelling results, the iron front is expected to affect only a few centimetres, even after hundreds of thousands of years (Wersin et al. 2007).

As a result of continuing land uplift prior to any future glaciation, the saline water currently at repository depth will start to be replaced by brackish, sulphate-rich and fresher groundwater currently present at shallower depths, and the driving forces for

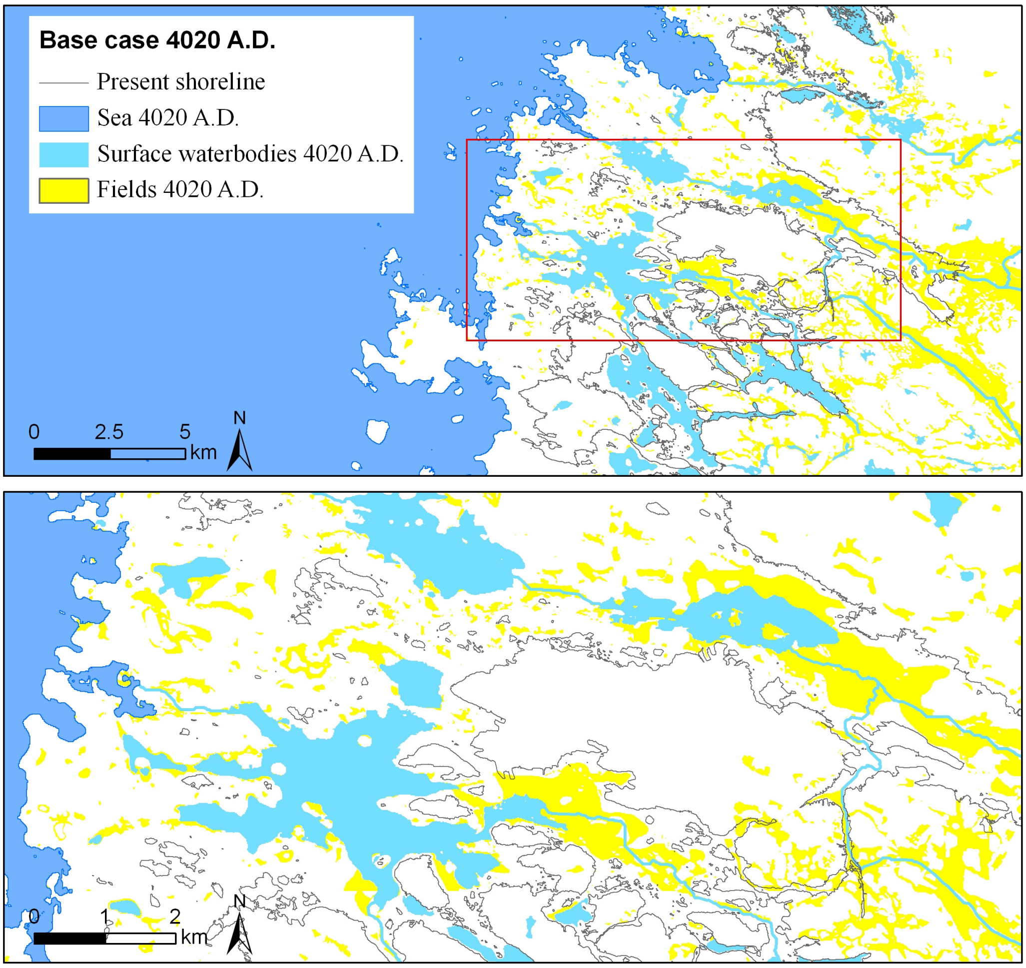


Figure 5-4. *Predicted evolution of the sea level, croplands and surface water bodies over the next two thousand years, the present coastline is shown as a grey line (Ikonen et al. 2010), map layout by Jani Helin, Posiva Oy.*

groundwater flow will also change. The changes in the salinity of the groundwater will affect the swelling pressure of the buffer, with lower salinities giving higher swelling pressures. The land uplift rate is expected to vary little over the next few centuries, but will decrease significantly within the next few thousand years.

The copper canisters will continue to slowly corrode. Given the expected anoxic conditions at the canister surface, the only significant corrosive agent will be sulphide that diffuses through the buffer from the rock. Because of the small diffusive flux of sulphide that is expected to reach the canister surface, the rate of copper corrosion will be very low.

5.4 Timing and impact of major climate changes

Climatic evolution is subject to significant uncertainties. In particular, climatic warming due to anthropogenic emissions, especially greenhouse gases, may prolong temperate conditions and considerably delay the future formation of permafrost and ice sheets at Olkiluoto (the Moderate Anthropogenic Emissions scenario described in Section 4.2.2).

Prolonged temperate conditions without glaciation will possibly result in a longer period of infiltration of dilute surface water, or even seawater (if sufficient sea-level rise takes place), towards repository level compared with the case in which no significant climatic warming due to such anthropogenic effects takes place. However, the infiltration of surface water is not expected to significantly reduce the ionic strength of the groundwater at repository depth, because of organic activity in the soil layers, interaction of slowly infiltrating surface water with rock minerals and mixing with deeper, more saline groundwater layers.

In all climatic scenarios, permafrost and frozen ground will eventually form at the site, although the timing is uncertain. Permafrost and frozen ground will have only a minor influence on hydro-mechanical pressure conditions at repository depth, possibly leading to a more stagnant flow pattern. However, the occurrence of ice sheets could result in transient and localised increases in the water pressures and hydraulic gradients in the subsurface. A particular concern is that the retreat and melting of ice sheets could result in large volumes of meltwater being forced into deeper parts of the bedrock. When an ice sheet is present, seismic activity, which is in any case low in the Olkiluoto region, will be further repressed. Post-glacial earthquakes may, however, occur following the retreat of the ice sheet, giving rise to stress changes in the rock that could trigger shear movements on smaller-scale fractures that intersect the deposition drifts, tunnels and holes.

The repository is designed to tolerate these processes associated with major climate change, and, in the current base scenario, the radionuclides associated with spent fuel remain totally contained within the canisters over a prolonged time frame. Some of the potentially disruptive events associated with climate change, however, give rise to repository assessment scenarios involving canister failure and radionuclide release, as described in Chapter 6.

6 FORMULATION OF ASSESSMENT SCENARIOS

The present chapter deals with the formulation of assessment scenarios, including repository assessment scenarios and dose assessment scenarios (Fig. 6-1). Human intrusion scenarios, which are also classified as assessment scenarios, will be formulated and analysed in future safety studies.

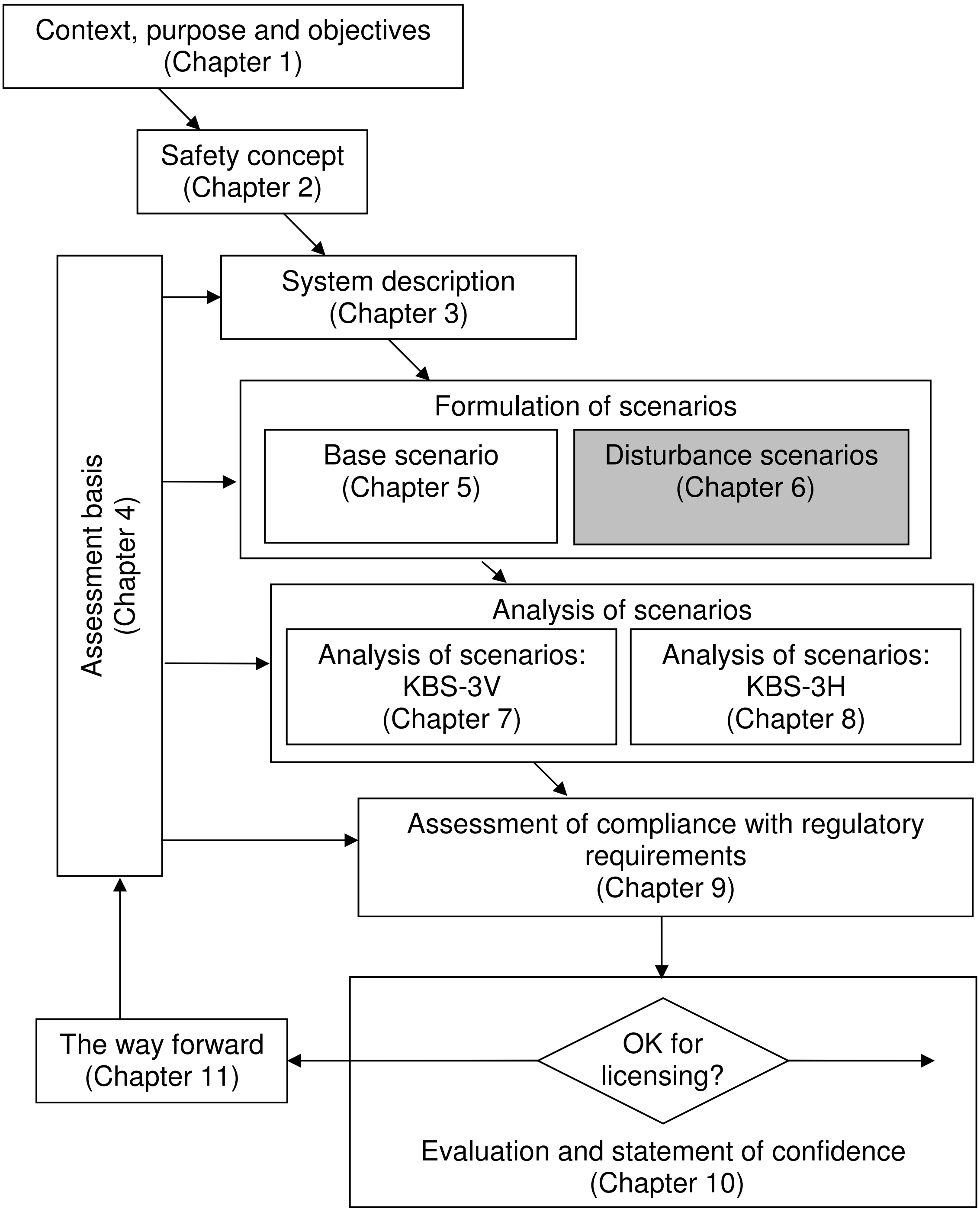


Figure 6-1. *The present chapter in the context of the safety case summary report.*

The chapter is structured as follows:

- Section 6.1 describes a general methodology for the formulation of assessment scenarios;
- Section 6.2 identifies the specific repository assessment scenarios considered in the 2009 KBS-3V safety analysis and in the KBS-3H safety assessment;
- Section 6.3 identifies the dose assessment scenarios considered in these studies.

6.1 Methodology for formulation of assessment scenarios

The current methodology to formulate repository assessment scenarios is as follows:

1. consider the features of the disposal system and, in the case of the main repository components, performance targets or target properties required for each of them to fulfil their respective safety functions;
2. develop understanding of the system and its evolution - with a focus on the most important events and processes that are likely to affect it;
3. identify less likely or uncertain events and processes that could significantly affect disposal system evolution - in the case of the main repository components, these are uncertain events and processes that could lead to failure to achieve performance targets or target properties and hence, potentially, to loss of, or significant perturbation to, the safety functions;
4. consider if and when the occurrence of such events and processes is plausible;
5. consider the implications of loss of, or significant perturbation to, one system feature or safety function on the others and also the possibility of more than one system feature or safety function being independently perturbed;
6. based on the above, identify (a), plausible descriptions of the evolution of safety functions over time that lead to radionuclide release from the repository - the repository assessment scenarios, and (b), plausible descriptions of the development of the surface environment, the migration paths for radionuclides to the surface environment and the usage (humans and other biota) of the surface environment that may affect the potential fate of radionuclides - the dose assessment scenarios.

The resulting scenarios are cross-checked to ensure inclusion of all scenarios required by Finnish regulations (YVL E.5): see, for example, the requirement to consider human intrusion scenarios noted above. Furthermore, in the KBS-3H safety assessment, a detailed comparison of the scenarios and calculation cases considered with those of TILA-99 and SR-Can was also used to confirm that there were no omissions or gaps, apart from where limitations related to the scope of the assessment mean that the treatment of some uncertainties was put aside (Neall et al. 2007).

Steps 1 and 2 relate to material described in earlier chapters. Steps 3-6 are covered in the following sections of the present chapter.

6.2 Repository assessment scenarios

Repository assessment scenarios explore the consequences of various uncertain features and perturbing processes that could potentially lead to canister failure, or significantly degrade the capacity of the repository to limit radionuclide transport in the event of canister failure.

In accordance with the methodology outlined in Section 6.1, repository assessment scenarios arise from less likely or uncertain events and processes that have the potential to compromise the capacity of the repository to meet its performance targets, or to significantly perturb the target properties of the bedrock. Scenarios arising from uncertain features and processes internal to the repository are considered in Section 6.2.1. Section 6.2.2 considers scenarios arising from uncertain external events. A summary of repository assessment scenarios is given in Section 6.2.3. The various uncertain events and processes identified in these sections as leading to scenarios are described in more detail in the KBS-3V Process Report (Miller & Marcos 2007), KBS-3V Evolution Report (Pastina & Hellä 2006) and KBS-3H Evolution Report (Smith et al. 2007b). The formulation of scenarios will, in future, be addressed in the Formulation of Scenarios Report (Fig. 1-3).

It should be noted that, in deriving repository assessment scenarios, the methodology described in Section 6.1 has so far been systematically applied by Posiva only in the KBS-3H safety assessment, using the "safety function indicator criteria" approach developed by SKB in its most recent safety assessment SR-Can (SKB 2006a). Tentative performance targets and target properties are presented in TKS-2009 (Posiva 2009a, Section 6.1.4). However, the repository scenarios formulated in the KBS-3H safety assessment largely overlap with the scenarios considered in the 2009 safety analysis of a KBS-3V repository, which were defined following an approach documented in Miller & Marcos (2007).

6.2.1 Scenarios arising from internal features and processes

Uncertain features and processes arising internally within the repository that could potentially lead to canister failure, or degrade the capacity of the repository to limit radionuclide transport in the event of canister failure have been identified in Section 4.1.3. These comprise:

- the possible presence of penetrating and non-penetrating defects in the canisters or other defects that could lead to early releases;
- internal processes leading to missing, loss or redistribution of buffer mass, with consequences for copper corrosion and radionuclide transport;
- internal processes leading to perturbation of the buffer / rock interface, also with consequences for copper corrosion and radionuclide transport;
- gas generated internally within the canister; and
- criticality.

Presence of defects in the copper canister overpacks

The presence of non-detected penetrating defects or other defects that could lead to early canister failure cannot currently be excluded. The probability of occurrence of such defects is not yet quantified, but will be reduced by, for example, suitable weld inspection and quality control procedures (Raiko 2005). Current understanding is that minor defects could occur anywhere in the copper overpack of a canister, but significant defects are most likely to occur along welds and, in particular, at the seal of the canister top lid. Ongoing work aims to quantify, in terms of a probability, the low likelihood of a defective weld escaping detection (Posiva 2009a, Chapter 5). A performance target is that each canister has complete copper coverage over its entire surface for hundreds of thousands of years. Defective canister scenarios in which this target fails to be met are considered in the 2009 KBS-3V safety analysis and the KBS-3H safety assessment.

Processes leading to missing or lost buffer mass

Performance targets on buffer density set the range over which the buffer safety functions are expected to operate. One possible cause of failure to meet these targets is improper emplacement of the buffer, the likelihood of which will be reduced by implementing a suitable system of quality control. A programme for testing and demonstration buffer emplacement at ONKALO is starting in 2010. A plan for QA/QC to be applied during installation of buffer blocks will be drawn up in the course of the emplacement tests. Nonetheless, the possibility of improper emplacement cannot currently be excluded. Other internal phenomena could conceivably result in loss or redistribution of buffer mass over time, including, for example, erosion due to piping or other transient water flows during repository operations and early evolution, displacements of KBS-3H supercontainers and distance blocks caused by uneven buffer swelling. Adverse effects will be reduced or avoided, for example, by the application of appropriate rock suitability criteria for locating deposition holes. Furthermore, scoping calculations and more qualitative arguments indicate that these phenomena are unlikely to compromise the capacity of the buffer to meet its performance targets on buffer density (see Appendix B of the KBS-3H Evolution Report, Smith et al. 2007b). They may, however, have a minor impact on the diffusion coefficient of the buffer. Even if large amounts of bentonite are missing, or some of the bentonite is lost by erosion, the bentonite will swell and fill the empty space, but the resulting swelling pressure may be rather low, and advective conditions in the buffer may arise (Börgesson & Hernelind 2006b).

Scoping calculations have also been carried out addressing the impact of perturbations to the buffer diffusion coefficient and of the development of advective conditions in the buffer on canister corrosion due to reaction with sulphide in the groundwater (see again Appendix B of the KBS-3H Evolution Report, Smith et al. 2007b). Results indicate that it is the abundance and rate of transport of sulphide in the groundwater, rather than the transport resistance provided by the buffer, that determines the corrosion rate. Only in the case of a high groundwater sulphide concentration of 100 mg per litre being maintained over a prolonged period, e.g. by microbial activity, coupled with advective conditions in the buffer, is the canister lifetime reduced below about 100 000 years. Nevertheless, increased buffer diffusion coefficients or the establishment of advective conditions in the buffer could have some impact on radionuclide transport in the event of canister failure.

A scenario in which the canister fails due to disruptive events affecting the buffer is considered in the 2009 KBS-3V safety analysis, and several calculation cases with perturbed radionuclide transport properties in the buffer surrounding a canister with an initial penetrating defect are considered both in the KBS-3H safety assessment and KBS-3V safety analysis.

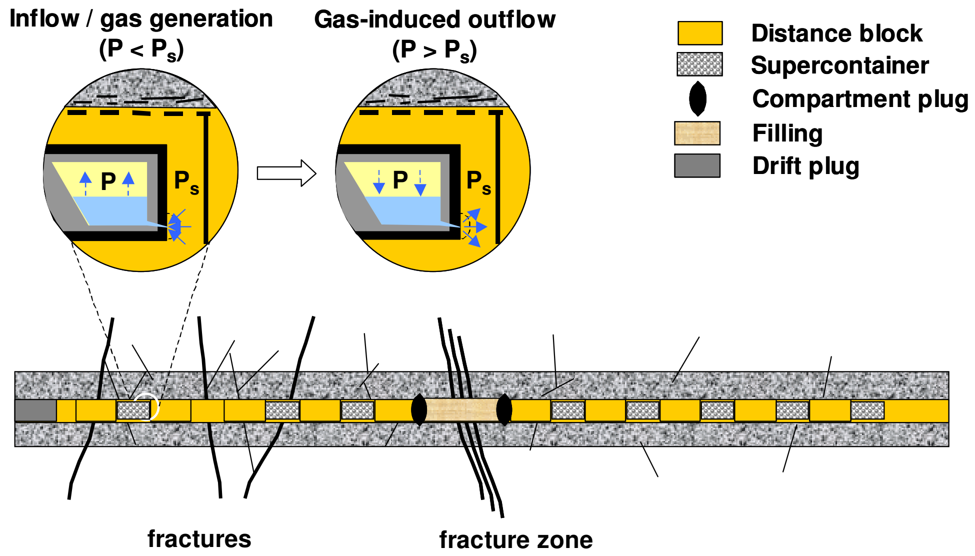
Processes leading to perturbation of the buffer/rock interface

The presence of an EDZ or the occurrence of stress- or thermally-induced rock spalling could lead to perturbed chemical and mass transfer conditions at the buffer/rock interface. There are also chemical phenomena that could affect conditions at the interface, such as interaction of the buffer with cement leachates and with Fe(II) from the corrosion of the KBS-3H supercontainer shells. The scoping calculations in Appendix B of the KBS-3H Evolution Report, Smith et al. 2007b) address the impact of a perturbed buffer/rock interface on canister corrosion, and, as noted above, indicate the effects to be minor except if high groundwater sulphide concentrations are maintained over a prolonged period. Nevertheless, a highly perturbed buffer/rock interface could have some impact on radionuclide transport in the event of canister failure. Increased mass transfer across the buffer/rock interface around a canister with an initial penetrating defect is considered in the KBS-3H safety assessment. Calculation cases in which the flow at the interface is varied are also analysed in the 2009 KBS-3V safety analysis.

Gas generated internally within a failed canister

Significant volumes of gas will be generated by the corrosion of the cast iron canister insert following canister failure. Some of the C-14 released from the spent fuel can partition into a free gas phase and mix with corrosion-generated gas inside the canister, before being expelled once the gas pressure is sufficiently high.

Depending on the canister corrosion rate, on the rate of supply of water from the buffer, and on the location of the point of failure in the copper overpack, another possibility is that water enters the canister interior, dissolves radionuclides and is then forced out of the canister once the gas pressure exceeds the confining pressure of the bentonite buffer (Fig. 6-2). Results of scoping calculations that are presented in the KBS-3H Process Report (Gribi et al. 2007), but are also valid for KBS-3V, indicate that the most likely situation is that water entering the canister will be completely consumed by corrosion of the cast iron insert, and there will be no gas-induced displacement of contaminated water through the defect into the saturated bentonite. Furthermore, if canister failure is due to the presence of an initial defect in the weld, expulsion of contaminated water by gas is a more likely possibility for KBS-3H than for KBS-3V. This is because expulsion of contaminated water by gas requires the defect to be located at a low vertical position on the canister surface, such that gas lying above the water in the canister interior can force this water out through the defect. In the case of a KBS-3V repository, the canisters will be emplaced vertically upright, such that the weld is near the top. Thus, expulsion of contaminated water by gas requires a canister to be emplaced upside down in the deposition hole. The possibility of such a situation arising is minimised in the planned operational procedure (see Tanskanen & Palmu 2004).



Note: the amount of free gas (light yellow) within the canister is changed by a number of different processes (gas generation, advection and diffusion of dissolved gases, dissolution / degassing).

Figure 6-2. *Conceptual model for transport of water and gas into and out of a KBS-3H canister with an initial penetrating defect (after Gribi et al. 2007). It is emphasised that an actual defect may be located anywhere around the canister weld, but that expulsion of contaminated water by gas can only occur if the defect is located in a low vertical position, as depicted in the figure.*

These processes are addressed in a scenario analysed in the 2009 KBS-3V safety analysis, and in calculation cases of the KBS-3H safety assessment.

Criticality

Based on the investigations carried out to date, and assuming credit for burnup to be taken, sustained induced fission (criticality) is not expected to occur in any of the types of canisters and for any of fuels that will be disposed of in a Finnish repository (see Anttila 2005 and Section 8.1.1 of Smith et al. 2007b) Scenarios including criticality have not been considered quantitatively in either of the recent KBS-3V and KBS-3H safety analyses. The possibility of criticality is not, however, completely ruled out, and will be considered further in future safety studies.

6.2.2 Scenarios arising from external events

Potentially detrimental processes arising from events external to the repository and that could potentially lead to canister failure, or degrade the capacity of the repository to limit radionuclide transport in the event of canister failure, have also been identified in Section 4.1.3. They comprise:

- buffer freezing;
- canister failure due to isostatic load;
- migration of oxygen to repository depth;
- loss of buffer due to exposure to glacial meltwater; and

- canister failure due to rock shear.

Most are related to major climate change, although canister failure due to rock shear could potentially occur, though with low probability, at any time.

Buffer freezing

At Olkiluoto, according to present knowledge based on past glaciations, any future permafrost layer is not expected to penetrate more than 180 metres below ground, indicating that freezing of the buffer is not to be expected (Hartikainen 2006). There remains the unlikely possibility that conditions at Olkiluoto could in the future differ significantly compared with those during the past glaciations and lead to buffer freezing. This possibility is receiving consideration in ongoing studies. It should be noted that heat transfer below frozen ground is predominantly by diffusion. There is thus a quadratic relationship between permafrost penetration depth and the duration of the permafrost: a factor of two increase in penetration depth requires a factor of four increase in duration. The potential consequences of buffer freezing and thawing on the buffer itself and on other system components, including canister integrity, are also being investigated. Recent laboratory experiments conducted at -18°C have demonstrated that the swelling pressure of compacted bentonite recovers to within 84 to 98% of its initial equilibrium value after several freeze/thaw cycles (Posiva 2009a, Section 6.5.4.2). In safety analyses to date, any damage to the buffer safety functions that freezing could cause has been implicitly taken into account in calculation cases where the parameters of the buffer have been cautiously selected to account for buffer disturbances.

Canister failure due to isostatic load

The highest isostatic loads on the canister will occur in association with future glaciations. Future ice sheets are not, however, expected to be thick enough to produce isostatic loads that would lead to canister failure, based of insert strength measurements, and this scenario has thus not been considered quantitatively in safety analyses (see, e.g., Section 4.1.2 of the present report, Section 8.1.2 of Pastina & Hellä 2006 and Section 7.4.4 of Smith et al. 2007b).

Migration of oxygen to repository depth

A target property of the bedrock is that conditions should be reducing, with no dissolved oxygen. Glacial meltwater penetrating the bedrock might carry with it oxygen that could accelerate corrosion if it reached the surface of the canisters. However, the recent interpretation of hydrogeochemical and mineralogical site data gives no evidence for intrusion of oxygen to repository depth at Olkiluoto in the past (see Section 3.2.3 and Posiva 2009b). This can be attributed to the consumption of oxygen by microbially-mediated reactions and the interaction of oxygen with minerals in the upper parts of the rock. Even if oxygenated water were to reach repository depth, scoping calculations by Ahonen & Vieno (1994) indicate that canister failure by corrosion hypothetically requires exposure to this water to be maintained for at least 100 000 years. In the case of KBS-3H, an even longer exposure time might be required, due to the oxygen scavenging effects of iron in divalent form (Fe (II)) originating from the corrosion of the steel supercontainer shell. Oxygen penetration has not been considered as a potential cause of canister failure in safety analyses, although it has not been completely ruled out and will be evaluated further in ongoing studies. The 2009 KBS-3V safety analysis and the KBS-3H safety assessment have both considered the impact of glacial groundwater chemistry on radionuclide migration subsequent to canister failure by some other cause.

Loss of buffer due to exposure to glacial meltwater

If the smectite clays in the buffer come into contact with water of low ionic strength, it is possible that the clays will be suspended as colloids and transported away from the deposition holes or drifts in flowing groundwater. A further target property of bedrock is that groundwater has sufficiently high ionic strength to avoid this "chemical erosion" of the buffer. On the basis of modelling studies, a transient reduction in the ionic strength of groundwater at repository depth in association with glacial retreat and the penetration of dilute glacial meltwater into the bedrock is considered possible at Olkiluoto (Fig. 8-6 of Pastina & Hellä 2006). Recent progress in the development of understanding of chemical erosion has been reported, for example, in Liu et al. (2009a, b). Chemical erosion is also a key topic within the BENTO research programme (Posiva 2009a, Section 6.5.4.2). Canister failure subsequent to buffer loss by chemical erosion has been considered as a scenario in the 2009 KBS-3V safety analysis and in the KBS-3H safety assessment.

Canister failure due to rock shear

Finland in general and the area around Olkiluoto in particular, has good tectonic stability, which is reflected by limited historical seismicity (Fig. 3-5). Recorded earthquake magnitudes in Finland have never exceeded 5 on the Richter scale (e.g. Mäntyniemi & Ahjos 1990, Ahjos & Uski1992). The most recent earthquake of magnitude 4.9 dates from the 1880s (Mäntyniemi 2005). Furthermore, there is no evidence of post-glacial faults at the site (Lindberg 2006). The occurrence of large earthquakes in the future (over 5 on the Richter scale), cannot, however, be excluded, particularly in association with the retreat of ice sheets.

Major features that are likely to be most affected by such earthquakes will be avoided by repository layout (Section 3.4.1). Nevertheless, a large earthquake could trigger secondary shear movements on fractures intersecting deposition holes or drifts. These movements could lead to deformation of the bentonite buffer and to additional stresses being exerted on the canisters which, if sufficiently large, could lead to rupturing. The likelihood of such an event will be further reduced by applying rock suitability criteria. As described in Section 3.4.1, rock suitability criteria (RSC) are being developed that will reduce the probability that a canister is emplaced in deposition holes or at drift locations intersected by fractures that could potentially undergo damaging shear movements. According to a preliminary criterion, deposition holes that are intersected by one or more fractures intersecting a deposition hole perimeter, determined by the fact that it is visible in all walls, will not be considered suitable for canister emplacement. This "full perimeter intersection" (FPI) criterion is currently being evaluated and will be revised as appropriate. A preliminary estimate of the number of canister positions with an FPI fracture (assuming the criterion is not applied) has been presented by Hellä et al. 2009 (p. 95). Considering the different variants of the geological DFN model (Buoro et al. 2009) and the tunnel observations, this estimate varies from 4 to 24% of the potential canister positions. Appropriate criteria for a KBS-3H repository have yet to be developed.

In the case of a KBS-3H repository, sufficiently extensive mineral transformation of the buffer by iron / bentonite interaction and the associated loss of buffer plasticity could make the canister more vulnerable to failure by rock shear in the event of a large earthquake. However, since mineral transformation is only expected to affect a small part of the buffer near to its interface with the rock (Wersin et al. 2007), the capacity of the buffer to protect the canisters from small rock shear movements is expected to be maintained in the event of such transformation.

Canister failure by major rock shear movements (causing displacements > 10 cm) has been considered as a scenario in the 2009 KBS-3V safety analysis and in the KBS-3H safety assessment, as is also required according to Paragraph 3.16 of regulatory guide YVL E.5:

"Unlikely events induced by natural phenomena to be considered shall include a major rock movement in the vicinity of the repository".

6.2.3 Summary of repository assessment scenarios

Tables 6-1 and 6-2 summarise the internal perturbing phenomena and phenomena arising externally, as discussed in the previous sections of this chapter, related engineered barrier performance target or bedrock target property (as defined in Ch. 3), their inclusion or omission in the most recent KBS-3V safety analysis and in the KBS-3H safety assessment (Chapters 7 and 8) and the rationale for inclusion or omission.

The consequences of inadvertent human intrusion were not included in these safety analyses, but are discussed in Section 10.2.3 of the present report based on safety assessments of Swedish sites and in earlier safety analyses in Finland.

Table 6-1. *Perturbing phenomena arising internally within the repository, related engineered barrier performance targets or bedrock target properties, inclusion or omission in KBS-3V and KBS-3H safety analyses and the rationale for inclusion or omission. Performance targets and target properties from Posiva (2009a), Section 6.1.4.*

Perturbing phenomenon leading to …	Related performance target or target property	Treatment in safety analyses		Rationale / comments
		KBS-3V (Ch. 7)	KBS-3H (Ch. 8)	
… initial defect in canister copper coverage	Performance target is that copper shall completely cover the canister interior.	Defective canister scenarios DCS-I and DCS-II (Section 7.2).	Calculation cases for canister with initial penetrating defect (Section 8.2).	Presence of penetrating defects or other defects that could lead to early canister failure not currently excluded; probability not yet quantified.
… missing, loss or redistribution of buffer mass	All buffer performance targets related to density potentially affected	AD-II scenario, which assumes increased diffusion coefficients (Section 7.3.2), and also one specific calculation case (LhB Q t4) that assumes high corrosion rates in the DCS-I scenario.	Increased buffer diffusion coefficient considered in variant calculation case for the initial penetrating defect failure mode.	Extent of possible buffer density reduction not currently quantified. May lead to corrosion failure in less than 100 000 years only in the event of relatively high sulphide concentrations in groundwater.
… perturbed conditions at buffer / rock interface	Buffer shall ensure a tight contact with the host rock.	Within each of the scenarios, a range of calculation cases in which the flow at the interface was varied.	Increased mass transfer across buffer / rock interface considered in variant calculation cases for the initial penetrating defect failure mode.	Scoping calculations indicate only minor effects on canister corrosion.
… gas generated internally within a failed canister	No specific targets, but bedrock target transport resistance based on assumption of transport of dissolved radionuclides, rather than gas-mediated transport.	Assessment scenario AD-III (Section 7.3.2), in which radionuclide release is affected by repository-generated gas.	Possibility of gas-mediated radionuclide transport and of expulsion of contaminated water by gas considered in variant calculation cases for the initial penetrating defect failure mode.	Considered in both safety analyses, although expulsion of contaminated water by gas more likely for KBS-3H than for KBS-3V because of the position of the postulated defect.
.. criticality	Could, if it occurred, affect the capacity of the system to meet many of the performance targets.	Not included.		Not ruled out completely, there will be some work done in the future.

Table 6-2. *Processes due to external events, related engineered barrier performance targets or bedrock target properties, inclusion or omission in KBS-3V and KBS-3H safety analyses and the rationale for inclusion or omission. Performance targets and target properties from Posiva (2009a), Section 6.1.4.*

Infrequent or unlikely disruptive event	Related performance target or target property	Treatment in safety analyses	Treatment in safety analyses	Rational / comments
		KBS-3V (Ch. 7)	KBS-3H (Ch. 8)	
Buffer freezing.	No specific targets, but performance target values for buffer density and swelling pressure based on assumption of unfrozen buffer.	Not specifically included (see Section 6.2.2, above).		Ruled out in the safety analyses on the basis of current understanding, but further study is underway.
Canister failure due to isostatic load.	Performance target that canister shall withstand the expected isostatic mechanical loads. Canister shall have sufficient mechanical strength to ensure minimal probability of isostatic collapse for isostatic pressures of up to 45 MPa.	Not included.		Ruled out on the basis of insert strength measurements compared with the estimates of expected isostatic loads during future glaciations.
Oxygen migration to repository depth.	Target property of reducing conditions in bedrock with no dissolved oxygen.	Case RS3g illustrates consequences of rapid intrusion of oxygenated meltwater		Ruled out in KBS-3H on the basis of recent interpretation of hydrogeochemical site data.
Loss of buffer due to exposure to glacial meltwater.	Target properties of bedrock relating to low flow around deposition holes/supercontainers and to groundwater ionic strength, which should be sufficiently high to avoid chemical erosion. All buffer performance targets related to density potentially affected if chemical erosion occurs.	A specific calculation case (B Sh-Lh q) in the AD-II scenario (Section 7.3.2).	Calculation cases for canister failure due to copper corrosion, following buffer erosion due to an influx of glacial meltwater (Section 8.4-1).	Cannot currently be excluded; subject of ongoing research.

Infrequent or unlikely disruptive event	**Related performance target or target property**	**Treatment in safety analyses** **KBS-3V (Ch. 7)**	**Treatment in safety analyses** **KBS-3H (Ch. 8)**	**Rational / comments**
Canister failure due to rock shear.	Performance target that canister shall withstand the expected dynamic mechanical loads. Canister shall have sufficient mechanical strength to ensure rupture limit > maximum shear stress on the canister, corresponding to a 10 cm displacement with a velocity of 1.0 m/s in any direction in the deposition hole.	AD-I scenario (Section 7.3.1).	Calculation cases for canister failure due to rock shear (Section 8.4-2).	In spite of the application of rock suitability criteria, a few fractures with the potential to undergo damaging movements are likely to escape detection. Consideration required by the Finnish regulatory guide YVL E.5.

6.3 Dose assessment scenarios

Dose assessment scenarios explore the consequences of the main uncertain features, events and processes that potentially could lead to alternative development and usage of the surface environment and migration paths for radionuclides into it. A dose assessment base scenario has been defined, along with other dose assessment scenarios that are intended to illustrate the impact of specific uncertainties or uncertainties in combination on potential radiological consequences of geosphere releases to humans and to other biota. The approach to formulating dose assessment scenarios has also been developed to be consistent with international recommendations (ICRP 2000, 2007).

The key drivers for the dose assessment scenarios are climatic changes and land use. A specific dose assessment scenario combines assumptions regarding future climatic evolution and future land use (termed *lines of evolution,* below) with other assumptions regarding uncertain features, events and processes, such as the characteristics and habits of humans and other biota, future human activities (other than land use) and infrequent natural events, such as forest fires, exceptional storms, etc.

In dose assessment, a dose assessment base scenario is defined, along with a number of other scenarios, each based on a potential line of evolution for climate and land use. The potential future climate states and broad types of land use that are considered for the purposes of dose assessment are described in Section 6.3.1. The dose assessment base scenario is described in Section 6.3.2. The dose assessment base scenario is the only scenario that has been used to evaluate dose in the 2009 safety analysis of a KBS-3V repository and in the KBS-3H safety assessment. Other scenarios have been defined, however, though not yet applied in dose assessment, which address alternative lines of evolution for climate and land use. These are described briefly in Section 6.3.3.

6.3.1 Potential future climate states and broad types of land use

As noted above, climatic changes and changes in land use are key drivers in the formulation of dose assessment scenarios. The potential future climate states and broad types of land use that are being considered for the purposes of dose assessment are given in Table 6-3.

Table 6-3. *Key drivers in the formulation of dose assessment scenarios and the variants under consideration (variants marked in bold analysed in the 2009 safety analysis of a KBS-3V repository)****.***

Scenario driver	Descriptions and divisions into variants
Climate	**CL1: "Present climate"**
	- Unchanged climatic conditions during dose assessment time window
	- Sea-level displacement caused by post-glacial land uplift
	- Flora and fauna as present
	CL2 "Warmer climate"
	- Increased temperatures during dose assessment time window
	- Sea-level displacement caused by post-glacial land uplift, and global sea level changes consistent with a warmer climate
	- Changes in flora and fauna
Land use	**LU1: "Present land use"**
	- Present land use characteristics assumed (cultivation, forestry and demography)
	- Unchanged land use during dose assessment time window
	LU2: "Urbanisation
	- The site is developed into an urban area within the dose assessment time window
	LU3: "Wilderness"
	- The site is abandoned by humans within the dose assessment time window and left in its natural state (unsettled and uncultivated)

6.3.2 Dose assessment base scenario

In the dose assessment base scenario, the present climate, land use and characteristic and habits of humans and other biota remain unchanged during the time window for biosphere assessment. It is thus based on the line of evolution CL1-LU1, using the notation of Table 6-3.

6.3.3 Other dose assessment scenarios

Other dose assessment scenarios can be envisaged based on the following lines of evolution for climate and land use:

CL2-LU1 Warmer climate and present land use

CL1-LU2 Present climate and urbanisation of the site

CL2-LU2 As above, but with warmer climate

CL1-LU3 Present climate and abandonment of the site (wilderness)

CL2-LU3 As above, but with warmer climate

Changes in the available technology could potentially affect the land use. Changes in the fields of radiobiology and epidemiology (related to our understanding of radiation risks at low doses and dose rates) or medical science (related to our capability for treating cancer) could affect the dose assessment. However, these types of changes are not considered in the dose assessment, which is consistent with the Guide YVL E.5, where it is stated that human habits can be assumed to remain unchanged within the time window where dose constraints apply.

7 ANALYSIS OF SCENARIOS IN THE 2009 KBS-3V SAFETY ANALYSIS

This chapter (Fig. 7-1) presents a summary of the results of the 2009 safety analysis of a KBS-3V repository.

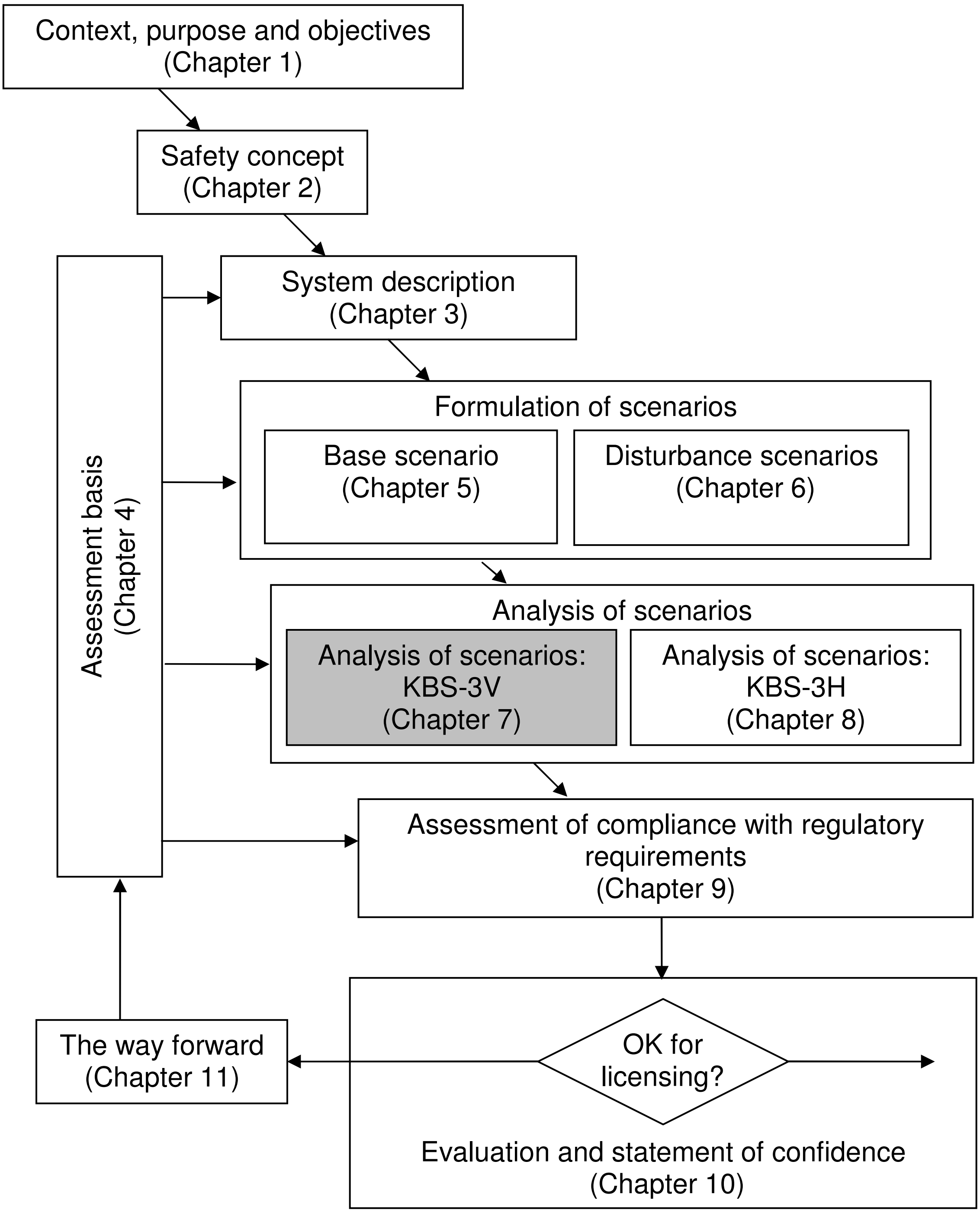


Figure 7-1. *The present chapter in the context of the safety case summary report.*

Since the near-field and geosphere modelling is separated from the biosphere assessment (Fig. 4-3), repository assessment scenarios and calculation cases are described first, followed by dose assessment scenarios and calculation cases. The chapter is structured as follows:

- Section 7.1 describes the scenarios and calculation cases considered in the safety analysis;
- Section 7.2 presents the releases from the geosphere evaluated in defective canister scenarios;
- Section 7.3 addresses additional repository assessment scenarios;
- Section 7.4 presents landscape modelling and the results of a subset of biosphere calculation cases that are classed as "realistic" (see Section 7.1.2); and
- Section 7.5 compares the results of the analyses with regulatory constraints.

The chapter summarises work presented in full in Nykyri et al. (2008) and in Hjerpe et al. 2010). In future, the analysis of scenarios will be described in the Analysis of Scenarios Report (Fig. 1-3).

7.1 Organisation of the safety analysis

7.1.1 Repository assessment scenarios

In the safety analysis of a KBS-3V repository (Nykyri et al. 2008), several repository assessment scenarios were formulated to cover the possible paths for system evolution involving radionuclide release, taking into account relevant perturbing phenomena discussed in Chapter 6. Each scenario comprises lines of evolution that are analysed or assessed with individual calculation cases, taking into account model and parameter uncertainties. The scenarios are divided into two groups.

Defective canister scenarios (Section 7.2):

- DCS-I: delayed penetrating defect – radionuclide release starting at 10 000 years after repository closure;
- DCS-II: early penetrating defect – groundwater in contact with spent fuel at repository closure.

Additional scenarios (Section 7.3):

- ADI-1: earthquake / rock shear: canister fails as a consequence of the sudden displacement of a fracture intersecting the deposition hole;
- AD-II: canister fails as a consequence of disruptive events affecting the buffer, e.g. misemplacement of the buffer, intrusion of dilute glacial melt water, etc;
- AD-III: gas expels water with instant release fraction and/or radionuclides in volatile form (C-14) from the canister and deposition hole; no credit is taken for any retention of radionuclides by the buffer and backfill.

In the defective canister scenarios (DCS), the containment capacity of the canister fails as a consequence of a defect in the copper shell, which is either a non-detected penetrating defect (hole) present from the outset that allows contact of water with spent fuel at repository closure (DCS-II), or a hole that develops over time from initially non-penetrating defects (e.g. thinner wall, non-penetrating tiny cracks, etc.) due to corrosion and allows contact of water with spent fuel and radionuclide release only after 10 000 years (DCS-I).

In the additional scenarios (AD), the canister containment capacity is lost due to internal phenomena or phenomena arising externally to the repository. In AD-I, the canister fails as a consequence of a rock shear displacement, as discussed in Section 6.2.5. In AD-II, accelerated corrosion damages the canister as a consequence of events affecting the buffer (e.g. initial poor emplacement of the buffer - Section 6.1.1 - or loss of buffer due to intrusion of glacial meltwater - Section 6.2.4). In AD-III, it is considered that, in a canister with a penetrating defect, gas is generated inside the canister due to corrosion of the iron insert and/or other metal parts. As described in Section 6.1.4, the gas generated will then be expelled and convey with it certain radionuclides in volatile form. It may also displace water that contains radionuclides originating mostly in the instant release fraction (IRF) of spent fuel; the IRF is discussed in Section 4.4.4.

Repository calculation cases for each repository assessment scenario are organised in tree structures that illustrate the main combinations of uncertainties considered, as shown in Figure 7-3 for the case of Defective Canister Scenario DCS-II, where an initial penetrating defect is present. A complete list of the analysed cases, as well as their classification as base and variant cases, sensitivity cases, “what-if” cases and supplementary cases, is given in Appendix 3 of Nykyri et al. (2008). The following sections present a selection of calculation cases and results. A full description of the tree methodology and a comprehensive description of all calculation cases considered in the KBS-3V safety analysis is given in Nykyri et al. (2008).

7.1.2 Dose assessment scenarios

As explained in Section 6.3, the dose assessment base scenario is the only dose assessment scenario that has been applied in the 2009 safety analysis of a KBS-3V repository, although other dose assessment scenarios have been identified and will be addressed in the PSAR. In the dose assessment base scenario, the present climate, land use and characteristic and habits of humans and other biota remain unchanged during the time window for biosphere assessment. As in the analysis of repository assessment scenarios, the dose assessment base scenario is analysed by means of individual calculation cases, termed biosphere calculation cases, taking into account model and parameter uncertainties. The biosphere calculation cases analysed in the 2009 safety assessment are shown in Table 7-1. A complete list of the analysed biosphere calculation cases, as well as their classification as realistic[11] cases, sensitivity cases, and “what-if” cases, is fully documented in Hjerpe et al. (2010).

The realistic and sensitivity dose assessment cases analysed in the 2009 safety assessment are summarised in Table 7-1. The pattern of radionuclide releases to the

[11] Here "realistic" means less conservative in terms of model assumptions and parameter values than in the other categories of cases.

surface environment will depend on from where in the repository the radionuclides originate. Three realistic biosphere calculation cases are defined in the 2009 safety analysis to address the effects of the location of a failed canister on the dose assessment. Each of these cases evaluates the fate of releases from a single failed canister, with the canister located in one of three different panels, denoted as Panels A, B and C[12], of the provisional repository layout shown in Figure 3-7. The biosphere objects to which releases originating from each of the panels are directed, and the distribution of the releases between these objects, are indicated in Figure 7-2.

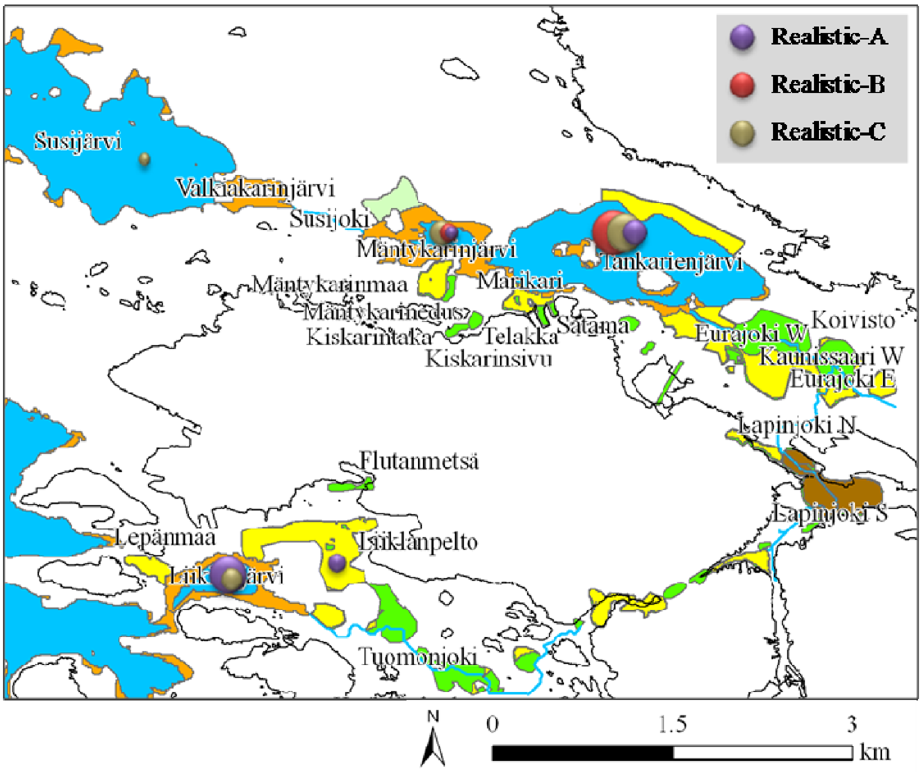


Figure 7-2. Schematic figure of the release patterns underpinning the realistic calculation cases in the analyses of dose assessment scenarios (grey line shows present coastline). The size of the circles represents the fraction of the releases directed to the biosphere object in question; see also 7-10 for a key to the different biosphere objects. Map by Jani Helin/Posiva Oy and Thomas Hjerpe/S&R Oy.

[12] Panels A, B and C correspond to Panels 1, 2 and 5, respectively, in Nykyri et al. (2008).

Table 7-1. Summary of biosphere calculation cases analysed in the 2009 safety assessment.

Biosphere calculation case	Aim
Realistic-A, B, and C	The cases considered to have an adequate level of conservatism for the considered scenario, in that the parameter values and assumptions are selected to ensure that the estimates of potential radiological consequences are cautious but still plausible and hence not unduly pessimistic
Sensitivity cases	
Developing surface environment	Evaluating the impact of changes in the stage of the development the surface environment at which the geosphere release reaches the biosphere
Release paths in the geosphere	Evaluating the impact of uncertainties in the distribution of release locations into the biosphere (caused by uncertainties in the initial location of the released radionuclides in the repository and the release paths to the biosphere)
Timing of geosphere releases	Evaluating the impact of uncertainties arising from when the geosphere release rate time series are introduced into the landscape model
Habits (humans)	Evaluating the impact of uncertainties in the way in which crops are irrigated.

A single dose assessment case can make use of the results of more than one repository assessment case, although not all possible combinations need be considered. The combinations of repository assessment case classes and dose assessment case classes considered in the 2009 safety analysis are shown in Table 7-2. To avoid excessive conservatism, the results of variant and sensitivity repository assessment cases are considered only in conjunction with the realistic biosphere assessment cases. Furthermore, neither the generally highly pessimistic “what-if” repository calculation cases nor the supplementary repository assessment cases and are analysed with biosphere calculation cases.

The number of dose assessment calculations to be carried out is also limited by the fact that only repository calculation cases that give rise to geosphere releases within the first 10 000 years are included in the dose assessment (the time window when the regulatory dose constraints are assumed to apply). In addition, as noted in Section 4.3.7, dose assessment using the landscape model is carried out only for a subset of those radionuclides for which geosphere releases are calculated. The results of the screening evaluation for the identification of radionuclides to be evaluated using the landscape model are given in Section 7.4, along with the annual doses evaluated using this model.

Table 7-2. *Treatment of KBS-3V repository calculation cases in the dose assessment.*

Repository calculation case classification	Biosphere calculation cases applied
Base case	Realistic and sensitivity cases
Variant cases	Realistic cases
Sensitivity cases	Realistic cases
"What-if" cases	None
Supplementary cases	None

7.2 Analysis of defective canister scenarios

7.2.1 The base repository calculation case

The base calculation case (Sh1) is a realisation of the DCS-II scenario (as shown in Fig. 7-3), in which an initial penetrating defect affects a single canister. The base calculation case serves as a reference for comparison with other calculation cases of this and other scenarios. Such comparisons illustrate the effects of specific model or parameter uncertainties. It should also be noted that many of these uncertainties are not specific to any particular canister failure mode. The assumption of an initial penetrating defect results in the earliest possible radiological impact, although not necessarily the largest impact, for each uncertainty considered.

In the base calculation case and in the majority of other calculation cases in the safety analysis (and also in the KBS-3H analysis), it has been assumed that radionuclides are released from a canister containing BWR fuel from OL1 or OL2 (although other fuel types have also been considered), with a burn-up of 40 MWd/kgU and an enrichment of 4.2%, which is at the high end of the currently expected range. A cooling time of 30 years has been conservatively assumed. Steady groundwater flow and geochemical conditions are assumed at all times. Groundwater conditions are taken to be reducing and dilute / brackish. Of the various groundwaters studied, dilute / brackish groundwater is considered closest in terms of total dissolved solids (TDS) to the expected undisturbed conditions at repository depth in the period up to 10 000 years in the future (Pastina & Hellä 2006). The geometry of the near-field model domain is the same as that shown in Figure 4-4. In all DCS calculation cases, it was conservatively assumed that a fracture intersects the deposition hole at a location that minimises the transport distance across the buffer between the defect and the fracture mouth (i.e. the centre plane of the fracture passes through the centre of the defect).

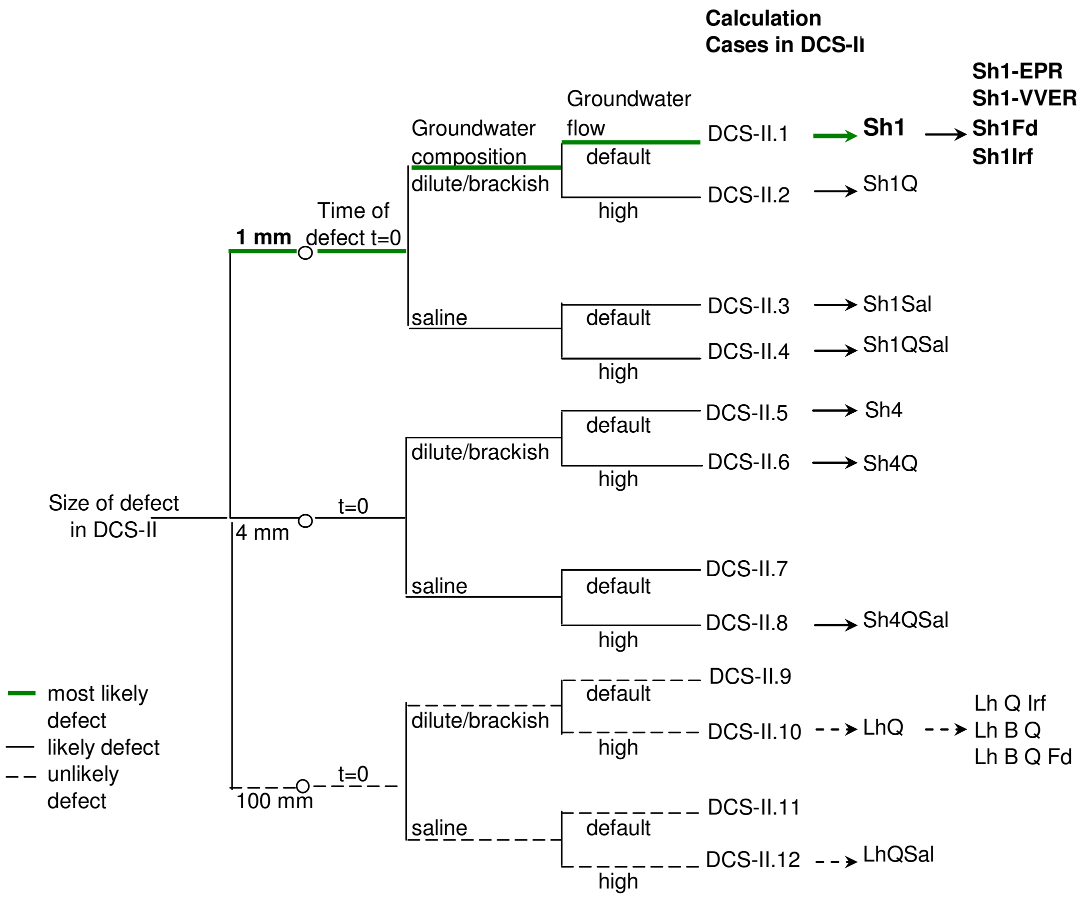


Figure 7-3. *The tree structure of the calculation cases in the Defective Canister Scenario DCS-II, where an initial penetrating defect is present in a canister (Nykyri et al. 2008, Fig. 6-1).* ***Sh1*** *is the base calculation case to which the results of other cases are compared. The abbreviation Sh is used in cases addressing a small defect (small hole). Lh is used in cases addressing a large defect (large hole). Other abbreviations are explained as required in later sections of the present chapter.*

The near-field and far-field (geosphere) release rates of some prominent radionuclides in the base calculation case are presented in Figure 7-4, showing the strong attenuation of shorter-lived or more sorbing radionuclides, such as Ni-59 and Sr-90, and the much weaker attenuation of longer-lived, weakly sorbing radionuclides, such as C-14, Cl-36 and I-129, which are cautiously treated as non-sorbing in the analysis. Figure 7-5 shows the evolution of a quantity termed the “overall release ratio” (or simply “release ratio” or “ratio”) and the contributions of the most important radionuclides. The overall release ratio is defined as the sum over all calculated radionuclides of nuclide-specific release ratios. A nuclide-specific release ratio is defined as the ratio of the activity release rate of a given radionuclide to the corresponding regulatory geo-bio flux constraint given in Guide YVL E.5. The overall release ratio is the parameter used to test compliance with regulations in the period after several thousand years. Longer-lived non-sorbing or weakly sorbing radionuclides give the greatest contributions to the overall release ratio.

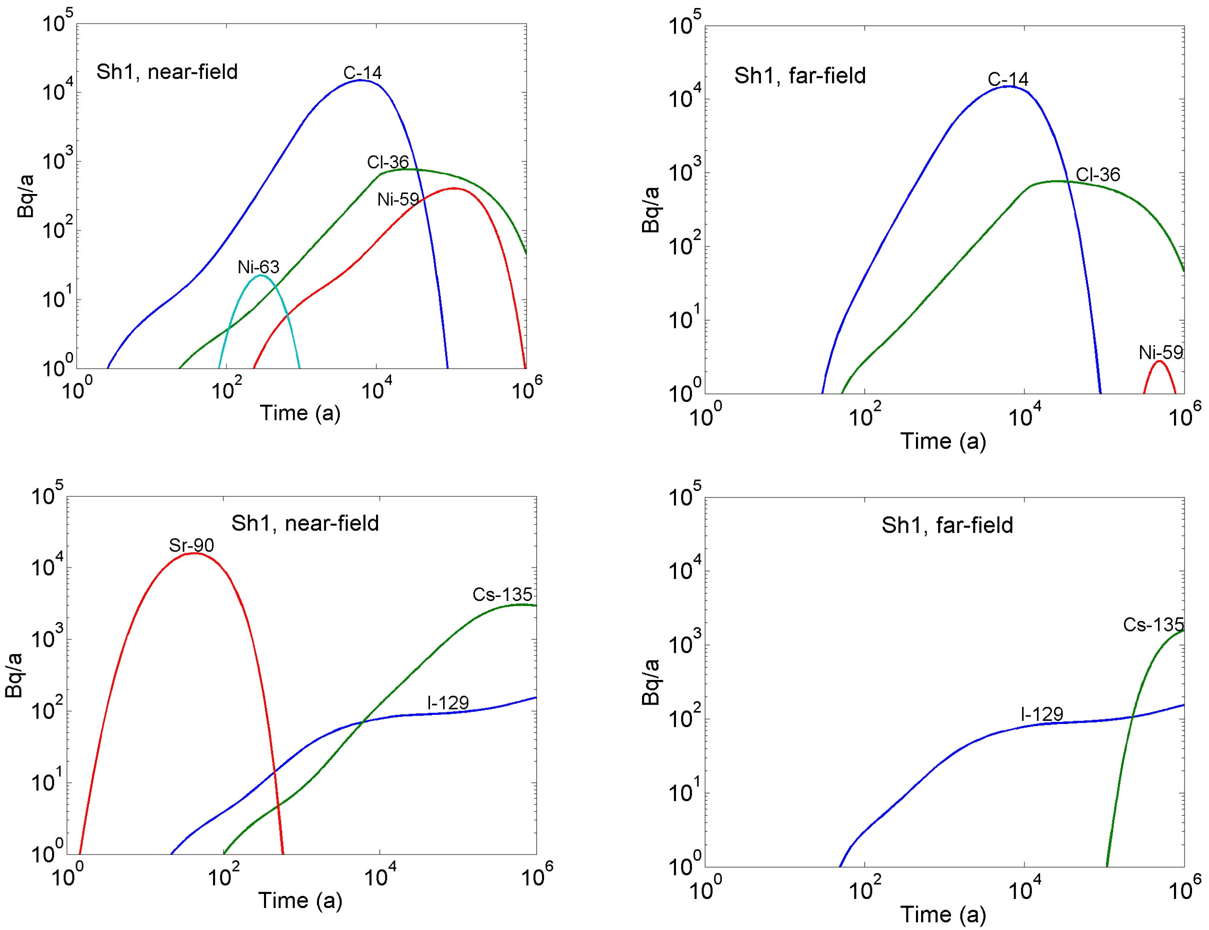


Figure 7-4. *Release rates from the near field (left) and far field (right) in calculation case Sh1 for two sets of radionuclides (Nykyri et al. 2008, Fig. 7-1)*

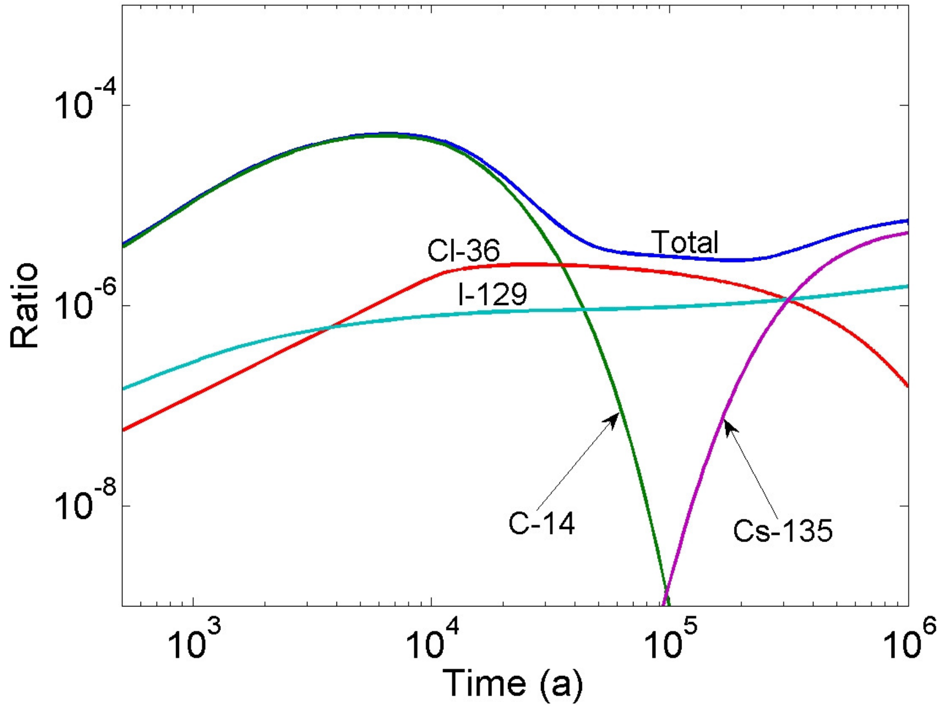


Figure 7-5. *KBS-3V safety analysis – base calculation case Sh1: Overall (total) release ratio as a function of time and the contributions of the most important radionuclides. The term "release ratio" is defined in the main text.*

The following paragraphs present the results of other calculation cases that illustrate the impact of some key uncertainties in near-field and geosphere release and transport modelling. A complete description of all repository calculation cases is given in Nykyri et al. (2008).

7.2.2 Effect of defect size

The size of any initial penetrating defect in the surface of a canister is highly uncertain, although in the safety analysis a the defect diameter was taken to be 1 mm, corresponding roughly to the maximum defect size that might escape detection using current non-destructive testing (NDT), and this was thus taken as the defect size in the base calculation case. In the small hole cases Sh4 and Sh4Q, the defect size was pessimistically increased to 4 mm. In the large hole cases Lh and LhQ, the transport resistance of the opening was, in effect, omitted by setting the defect size to a hypothetical large value of 100 mm. In cases Sh4Q and LhQ, and in the small hole case Sh1Q, the flow rate around the canister position is set to a relatively high value (a factor of 10 higher than in the base calculation case), which decreases the effect of geosphere retention. The overall release ratio in these high flow rate cases is shown in Figure 7-6. The effect of increasing the defect size from 1 mm to 4 mm is to increase the release ratio by a factor similar to the ratio of the defect areas (factor of 16). In the large hole case, the effect of the defect size increase is less pronounced, since other processes in the near field (solubilities and transport resistances) limit the releases.

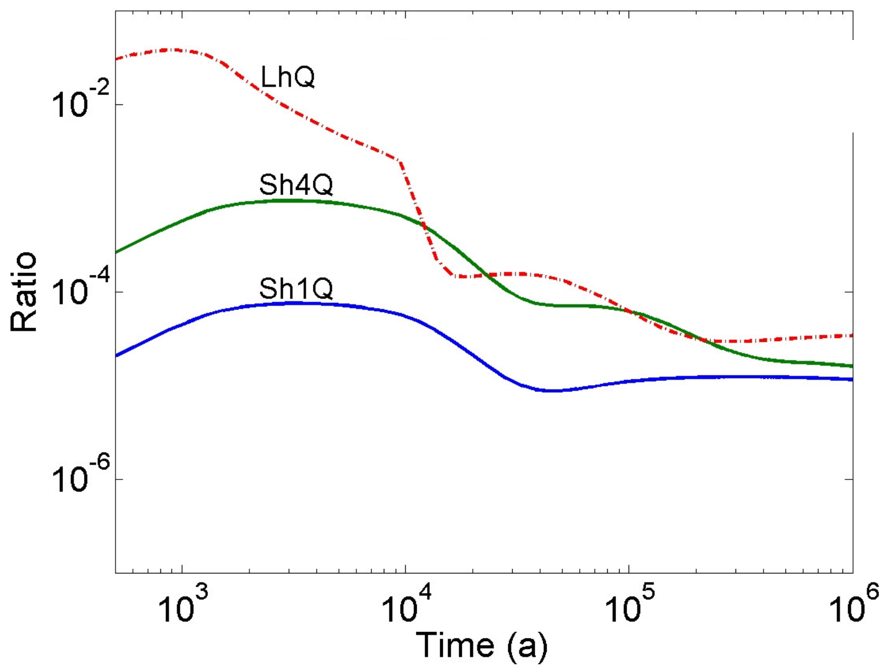


Figure 7-6. *The effect of defect size on release rate ratio in cases with high flow and dilute/ brackish groundwater chemistry. The suffix Q indicates a high flow calculation case. The only difference in the cases plotted is the defect diameters: Sh1 1 mm, Sh4 4 mm, and Lh 100 mm.*

7.2.3 Effect of flow rate

The slow rate of groundwater flow in the geosphere around the deposition holes limits both the release of radionuclides from the near field of a failed canister, and the rate at which these radionuclides are then transported through the geosphere. The flow around a canister position enters as an input parameter to the near-field release and transport model. The geosphere transport resistance, denoted as *WL/Q*[13], is the main hydrogeological input parameter of the geosphere transport model. The flow around canister positions and the geosphere transport resistance are both uncertain and variable in time and space. The effects of an increased flow rate around a canister position and a reduced geosphere transport resistance on near-field and far-field (geosphere) release rates are illustrated in the case of a canister with a 4 mm initial penetrating defect in Figure 7-7. In calculation case Sh4Q, the flow around a canister position is increased by a factor of 10 with respect to the base calculation case, and the geosphere transport resistance is reduced by the same factor. The results show that a higher flow rate is especially important for sorbing radionuclides such as Ni-59 and Cs-135. For weakly and non-sorbing radionuclides, such as I-129, the effect on the release is minor.

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[13] *W* is the width of the flow channel considered in geosphere transport modelling, *L* is the transport distance and *Q* is the flow rate in the channel.

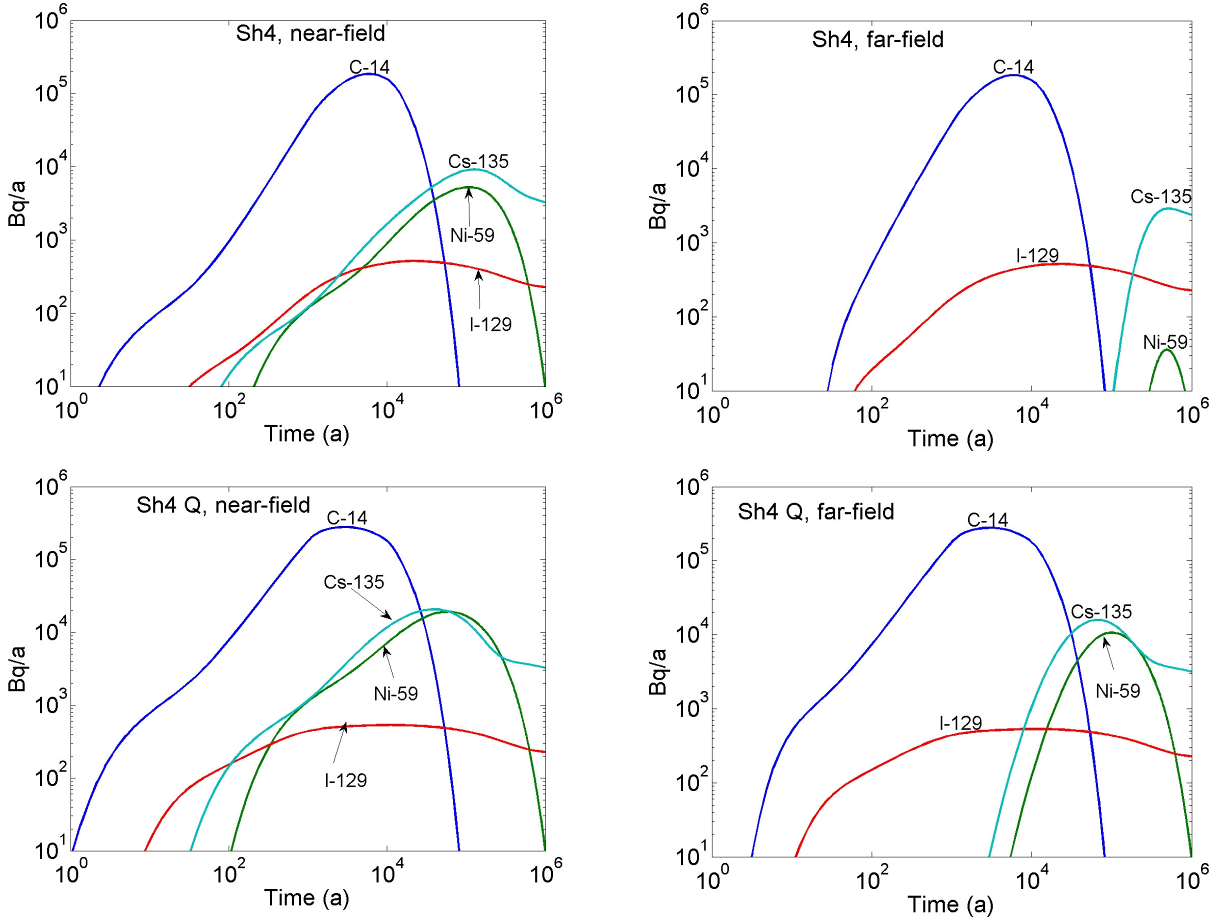


Figure 7-7. *Release rates from the near (left) and far field (right) in Sh4 (base-case flow) and Sh4Q (high flow) (Nykyri et al. 2008, Fig. 7-2).*

7.2.4 Effect of the instant release fraction

To assess the impact of uncertainties in the radionuclide-specific values used as instant release fraction (IRF), a hypothetical calculation case taking into account the instant release fraction alone - Lh Q Irf - is compared with case Lh Q in Figure 7-8. In these large hole / high flow cases (Lh Q), the IRF dominates the maximum release rates from the near field more than in the base calculation case, since the small size of the hole in the base calculation case and the limited flow around the deposition hole spread the release of the IRF from the near field over time.

C-14 and I-129 releases are shown in the figure. The four sources of C-14 can be clearly identified in the figure:

- the IRF;
- corrosion of Zircaloy cladding, leading to a release ending at 10 000 years;
- corrosion of other metal parts, leading to release ending at 1 000 years; and
- fuel matrix degradation, leading to release at a rate of 10^{-7} per year in the base calculation case.

Note that fuel matrix release declines rapidly after about 10 000 years due to the decay of the C-14 inventory (C-14 half life is 5 730 years).

In the case of I-129, it is assumed that its IRF is 5% of the total inventory, and this fraction dominates the maximum (peak) release rate. The rest of the I-129 inventory, which is in the fuel matrix, is released uniformly at the base case rate of 10^{-7} per year.

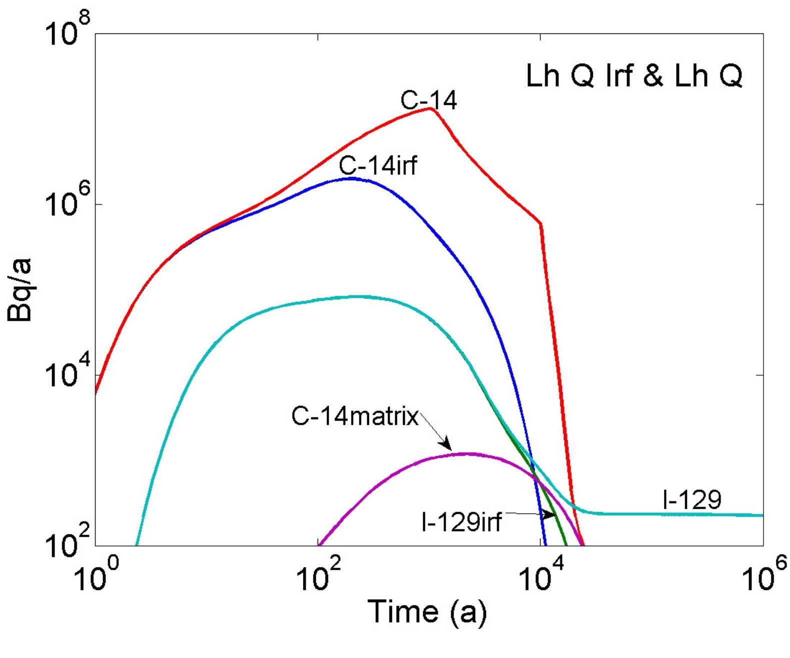


Figure 7-8. *Near-field release rates of C-14 and I-129 in case Lh Q Irf (IRF only) compared with case Lh Q (Nykyri et al. 2008, Fig. 7-3).*

7.3 Analysis of additional repository assessment scenarios

7.3.1 The rock shear/earthquake scenario (AD-I)

As described in Section 6.2.5, a large earthquake in the vicinity of the repository could lead to shear movements on fractures intersecting the deposition drifts, leading to the failure of unfavourably located canisters. Four calculation cases were studied (Fig. 7-9), in each of which a different canister failure time was postulated: RS1, with failure at 1 000 years, RS2, with failure at 10 000 years, and RS3 and RS3g, with failure at 70 000 years in the future. 70 000 years in the future will be the time of the next glacial retreat, assuming a repetition of the last glacial cycle. As noted in Sections 3.4.4 and 6.2.5, large earthquakes are most likely to occur at a site following the retreat of a future ice sheet.

In the four calculation cases, the location of the affected canister coincides with the location of the shearing fracture. Conservatively, no credit is taken for the delay between canister failure and the establishment of a radionuclide pathway from the canister interior to the buffer, and no credit is taken for any resistance to water ingress or radionuclide release from the failed canisters. The likely correlation between fracture

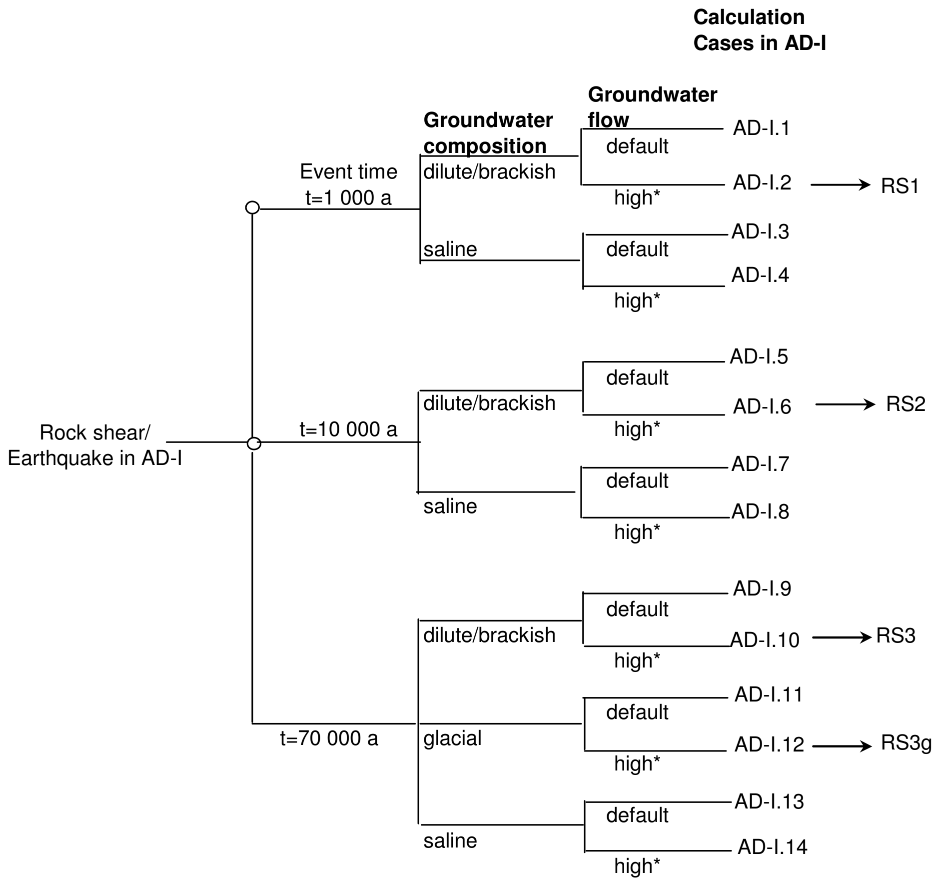


Figure 7-9. *Calculation cases for the Rock shear/Earthquake scenario AD-I. High flow rate applies at the fracture intersecting the deposition hole (Q_F) and in the geosphere (WL/Q), but not in the tunnel backfill and EDZ (Q_{DZ}, Q_{TDZ}) (Nykyri et al. 2008, Fig. 6-3).*

size and transmissivity means that, at deposition hole locations where rupture by rock shear occurs, the groundwater flow in the host rock is likely to be relatively high and transport resistance low. This likelihood is further increased by the effects of rock shear on the fracture. The flow in the fracture is therefore assumed to be high in all calculation cases for this scenario, leading to relatively rapid transport of radionuclides released from the buffer through the geosphere to the biosphere.

Table 7-3 compares the calculated overall release ratio maxima (as defined in Section 6.4.2) in the base calculation case for a canister with an initial penetrating defect (Sh1) with those for the rock shear/earthquake calculation cases. The nuclide-specific release ratio maxima for the three radionuclides giving the highest maxima in each case are also given. The release ratio maxima are more than two orders of magnitude higher in the rock shear/earthquake calculation cases compared with Sh1.

As in the base calculation case, the highest contribution to the maxima is from C-14 in cases RS1 and RS2, where the release from the canister starts at 1 000 or 10 000 years. In cases RS3 and RS3g, where the release event occurs later, the radioactive decay of C -14 (5 730 years half life) is effective in decreasing the maxima from this radionuclide and the highest contributions are from I-129 (1.6×10^7 years half life), followed by Cl-36 (3.0×10^5 years half life).

7.3.2 The disrupted buffer scenario (AD-II)

In scenario AD-II, a canister fails after some time due to corrosion, the buffer having a reduced capacity to protect the canister as a consequence, for example, of initial misemplacement (Section 6.1.1), or chemical erosion following intrusion of dilute glacial meltwater (Section 6.2.4). The calculation cases evaluated for this scenario are shown in Figure 7-10.

***Table 7-3**. Overall release ratio maxima and nuclide-specific release ratio maxima for the three most important radionuclides for the calculation cases in scenario AD-I (Rock shear/Earthquake scenario). The results of Sh1 are shown for comparison and the times of occurrence of the maxima are also given. The term "release ratio" is defined in the main text.*

Calculation case	t_{max} (a)	Overall release ratio maxima	1st Nuclide	Nuclide-specific release ratio maxima	2nd nuclide	Nuclide-specific release ratio maxima	3rd Nuclide	Nuclide-specific release ratio maxima
Sh1	**$6.5 \cdot 10^3$**	**$5.1 \cdot 10^{-5}$**	**C-14**	**$4.9 \cdot 10^{-5}$**	**Cs-135**	**$5.3 \cdot 10^{-6}$**	**Cl-36**	**$2.5 \cdot 10^{-6}$**
RS1	$1.7 \cdot 10^3$	$3.1 \cdot 10^{-2}$	C-14	$3.1 \cdot 10^{-2}$	I-129	$9.5 \cdot 10^{-4}$	Cl-36	$4.6 \cdot 10^{-4}$
RS2	$1.1 \cdot 10^4$	$1.1 \cdot 10^{-2}$	C-14	$1.0 \cdot 10^{-2}$	I-129	$9.4 \cdot 10^{-4}$	Cl-36	$4.5 \cdot 10^{-4}$
RS3	$7.1 \cdot 10^4$	$1.5 \cdot 10^{-3}$	I-129	$1.0 \cdot 10^{-3}$	Cl-36	$4.4 \cdot 10^{-4}$	Cs-135	$7.4 \cdot 10^{-5}$
RS3g	$7.1 \cdot 10^4$	$1.5 \cdot 10^{-3}$	I-129	$1.1 \cdot 10^{-3}$	Cl-36	$4.5 \cdot 10^{-4}$	Pa-231	$2.5 \cdot 10^{-4}$

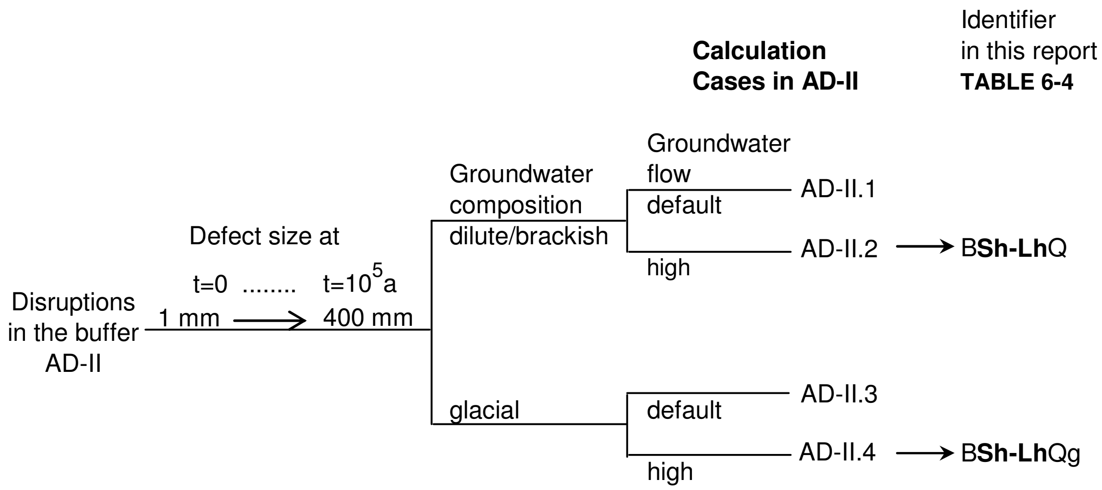


Figure 7-10. *Calculation cases in the Additional Scenario AD-II (disruptions in the buffer due to deviations in initial conditions and/or later events/processes) (Nykyri et al. 2008, Fig. 6-4).*

Calculation case BSh-LhQ assumes dilute/brackish groundwater, as in the base calculation case, and BSh-LhQg assumes glacial meltwater. In both cases, the initial small hole that is present in a single representative canister is assumed to lose its transport resistance at 100 000 years (modelled by increasing the defect diameter to a hypothetical value of 400 mm). After 100 000 years, no credit is taken for any further delay or attenuation of radionuclide releases provided by the perturbed buffer (advective conditions prevail).

7.3.3 The scenario of release affected by gas (AD-III)

In the scenario of release affected by gas (AD-III), the gas generated inside a canister with an initial penetrating defect is expelled, conveying with it C-14 in volatile form. Two calculation cases are considered. GASexG considers the transport of gaseous C-14 from the IRF with the repository-generated gas. Half of the IRF C-14 is assumed to be released to the geosphere at 900 years, when the gas pressure becomes large enough to create gas pathways in the buffer, with a more gradual release thereafter. Gas conveying C-14 is taken to be transported instantly to the geosphere, after which it dissolves in the large volume of water present in the host rock. GASexW considers the possibility that gas displaces water inside the canister containing radionuclides originating mostly in the IRF. Based on the most pessimistic case from a range of model calculations of the fate of water / vapour / gas and radionuclides described in Section 2.5 of the KBS-3H Process Report (Gribi et al. 2007), the gas-driven water pulse is taken to begin at 2 800 years after deposition and lasts for 1 300 years (the analysis is not specific to the orientation of the canister). Again, a simplified modelling approach is adopted whereby expelled contaminated water is conveyed instantly from the canister interior to the geosphere.

Table 7-4 compares the calculated overall release ratio maxima in the base calculation case for a canister with an initial penetrating defect (Sh1) with calculation case BSh-LhQ of the AD-II scenario and calculation cases BSh-LhQg, GASexW and GASexG of the AD-III scenario. The nuclide-specific release ratio maxima for the three radionuclides giving the highest maxima in each case are also given. The overall release ratio maxima are one to two orders of magnitude higher in the AD-II and AD-III scenarios compared with Sh1.

7.4 Analysis of the dose assessment base scenario

This section presents the results of landscape modelling and the assessment of radiological consequences for the dose assessment base scenario. The results presented only address the realistic biosphere calculation cases for this scenario, as defined in Section 7.1.2. Results from the analysis of sensitivity biosphere calculation cases are presented in Hjerpe et al. (2010).

7.4.1 The landscape model

The landscape model is presented in detail in Hjerpe & Broed (2010). Figure 7-11 gives a schematic illustration of the landscape model at the end of the biosphere assessment time window (year 12 020). The present-day coastline is shown as a grey line and it can be seen from the figure that the major part of the objects is currently under sea. The total number of biosphere objects in the landscape model is 70, and these contain between them 166 interconnected sub-objects; the distributions of ecosystem types of the sub-objects are:

24 forest sub-objects, 19 wetland sub-objects, 15 cropland sub-objects, 11 lake sub-objects, 29 river sub-objects and 68 coast sub-objects.

Table 7-4. *Overall release ratio maxima and nuclide-specific release ratio maxima for the three most important radionuclides for the calculation cases in scenarios AD-II and AD-III. The results of Sh1 are shown for comparison and the times of occurrence of the maxima are also given. The term "release ratio" is defined in the main text.*

Calculation case	t_{max} (a)	Overall release ratio maxima	1st Nuclide	Nuclide-specific release ratio maxima	2nd nuclide	Nuclide-specific release ratio maxima	3rd Nuclide	Nuclide-specific release ratio maxima
Sh1	**$6.5 \cdot 10^{3}$**	**$5.1 \cdot 10^{-5}$**	**C-14**	**$4.9 \cdot 10^{-5}$**	**Cs-135**	**$5.3 \cdot 10^{-6}$**	**Cl-36**	**$2.5 \cdot 10^{-6}$**
BSh-LhQ	$1.0 \cdot 10^{5}$	$1.5 \cdot 10^{-3}$	Cl-36	$1.0 \cdot 10^{-3}$	I-129	$4.6 \cdot 10^{-4}$	Cs-135	$1.3 \cdot 10^{-4}$
BSh-LhQg	$1.0 \cdot 10^{5}$	$1.6 \cdot 10^{-3}$	Cl-36	$1.1 \cdot 10^{-3}$	I-129	$4.9 \cdot 10^{-4}$	Cs-135	$1.3 \cdot 10^{-4}$
GASexW	$3.4 \cdot 10^{3}$	$4.6 \cdot 10^{-3}$	C-14	$3.4 \cdot 10^{-3}$	I-129	$9.4 \cdot 10^{-4}$	Cl-36	$2.7 \cdot 10^{-4}$
GASexG	$1.4 \cdot 10^{3}$	$3.0 \cdot 10^{-3}$	C-14	$3.0 \cdot 10^{-3}$	-	-	-	-

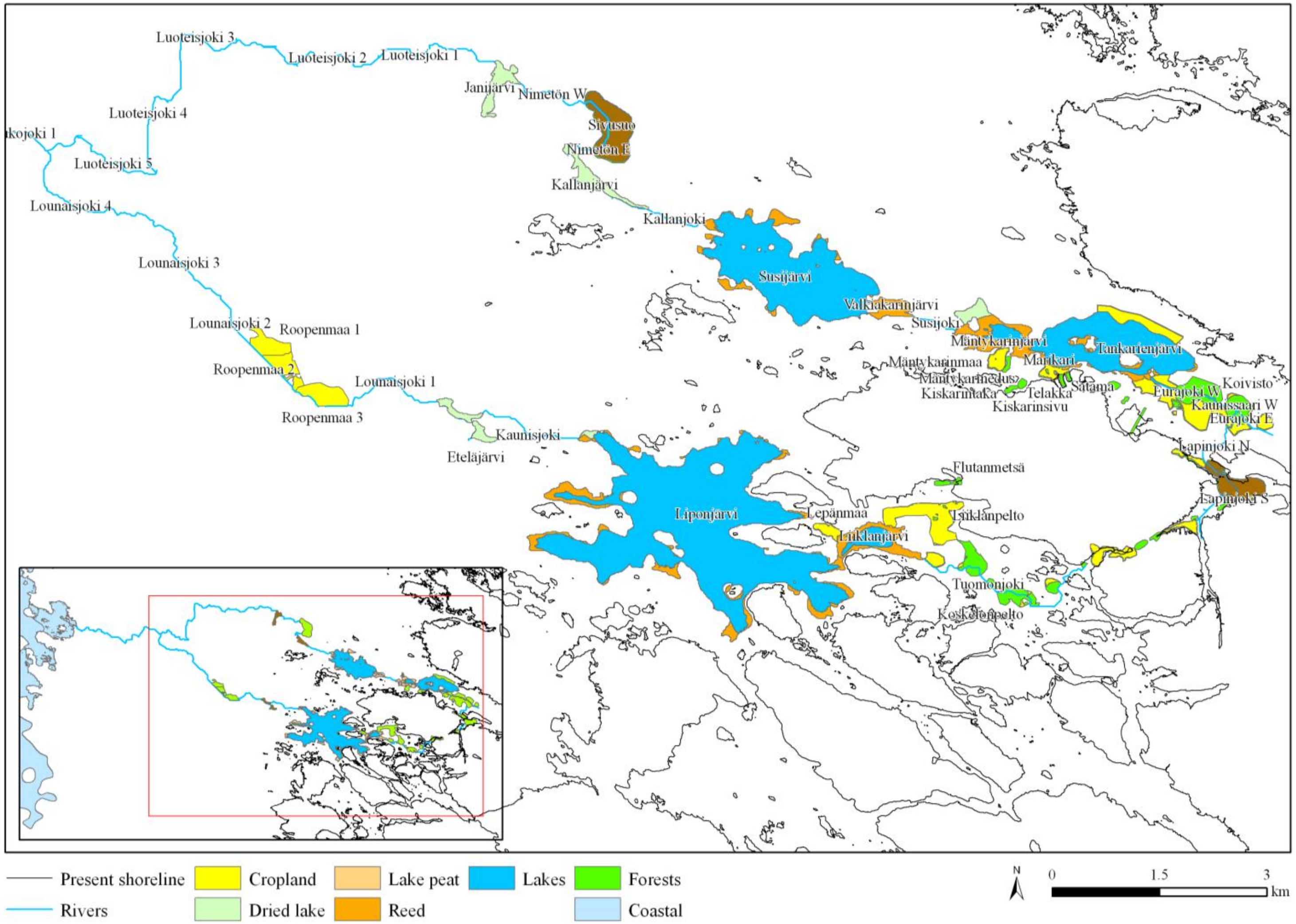


***Figure 7-11**. Schematic figure of the landscape model at year 12 020 (grey line shows present coastline). Map by Jani Helin/Posiva Oy.*

7.4.2 Screening evaluation

As described in Section 4.3.7, a 3-tiered approach to radionuclide screening has been applied in dose assessment. Those radionuclides with insignificant radiological consequences for humans and other biota are screened out in Tiers 1 and 2, and only those radionuclides that contribute significantly to dose are considered in the landscape model, which represents Tier 3.

The screening evaluation has been carried out for geosphere releases evaluated in those repository calculation cases for which biosphere calculation cases have also been applied[14] (Table 7-2). Of the about 40 radionuclides in the geosphere releases, radionuclides that are not screened out in Tiers 1 and 2 for any of the KBS-3V repository calculation cases considered are:

- C-14, Cl-36 and I-129.

Additional radionuclides that are not screened out in Tiers 1 and 2 for at least one of the repository calculation cases considered are:

- Ni-59, Se-79, Sr-90, Mo-93, Nb-94, Pd-107, Sn-126 and Cs-135.

7.4.3 Annual doses to humans

As described in Section 4.4, Guide YVL E.5 specifies constraints on the annual dose for the most exposed individuals and also for larger groups of people who may be exposed to radioactive releases. Thus, the annual landscape dose maxima to representative persons for the most exposed group and other exposed people, E_{group} and E_{pop}, have been calculated. The calculations are for 27 combinations of the three realistic biosphere calculation cases, as defined in Section 7.1.2, and the 9 repository calculation cases, 7 or which addressed releases from a small hole of 1 mm diameter, in the canister:

- Sh1: base calculation case - BWR fuel type; dilute/brackish groundwater;
- Sh1-EPR: considers EPR fuel type;
- Sh1-VVER: considers VVER fuel type;
- Sh1-IRF: considers IRF release only;
- Sh1Q: considers increased groundwater flow rate;
- Sh1Sal: considers saline groundwater; and
- Sh1Fd: considers increased fuel dissolution rate;

and 2 of which addressed a larger 4 mm diameter hole:

- Sh4; identical to base case, except for larger hole diameter; and
- Sh4Q: considers increased groundwater flow rate.

---

[14] A few repository calculation cases from the KBS-3H safety studies (PD-BC and PD-EXPELL) were also included in the screening evaluation. The results are then more general, and can be used to identify key radionuclides for a KBS 3 repository, regardless of design alternative.

E_{group} in these calculations ranges from about 5×10^{-7} to 3×10^{-5} mSv and E_{pop} ranges from about 10^{-8} to 5×10^{-6} mSv. The repository calculation case resulting in highest doses is Sh4 Q, both for E_{group} and E_{pop}.

Selected results for the combination of the realistic biosphere calculation cases together with (i), the base repository calculation case (Sh1, as defined in Section 7.2.1) and (ii), the repository case resulting in highest doses (Sh4 Q) are presented in Table 7-5 and 7-6 and in Figure 7-12.

Table 7-5. *Annual landscape dose maxima to a representative person for the most exposed group, (E_{group}), the year the maxima occur, the contribution to the dose maxima from different exposure pathways, and contributions from dominating radionuclides for selected cases (after table 7-7 in Hjerpe et al. 2010).*

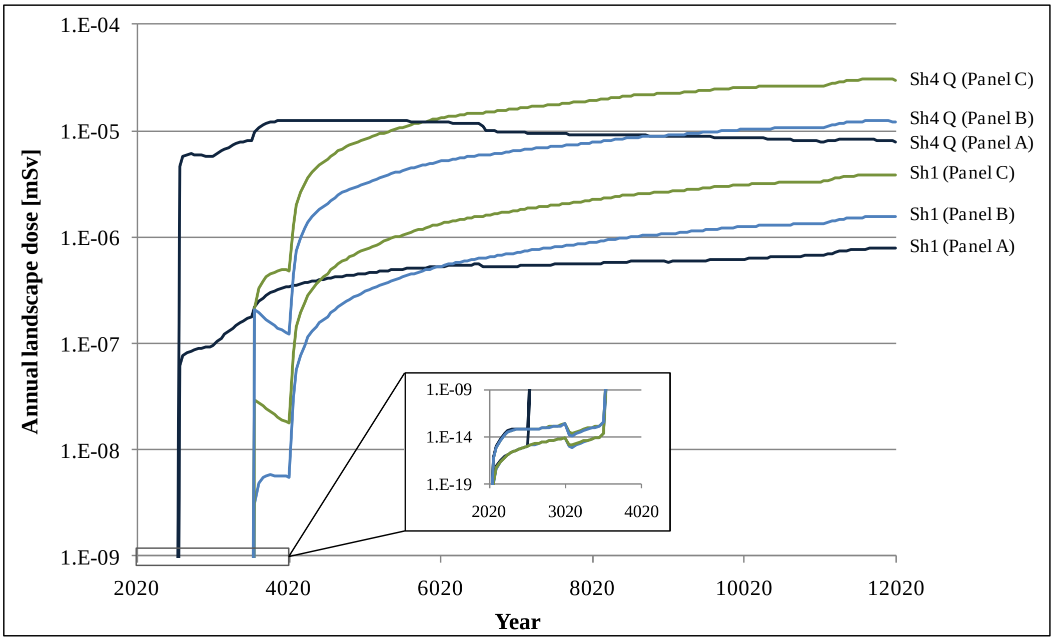
Case	**E_{group} [mSv]**	**Year**	**FI [%]**	**WI [%]**	**I-EE [%]**	**C-14**	**Cl-36**	**I-129**
Panel A								
Sh1	7.9×10^{-7}	11 920	94	6	0	0	46	54
Sh4 Q	1.4×10^{-5}	6 570	96	4	0	99	0	0
Panel B								
Sh1	1.6×10^{-6}	11 870	99	1	0	0	42	58
Sh4 Q	1.2×10^{-5}	11 820	99	1	0	0	47	53
Panel C								
Sh1	3.9×10^{-6}	11 870	100	0	0	0	41	59
Sh4 Q	3.1×10^{-5}	11 820	100	0	0	0	45	55

FI – contribution to the dose maxima from food ingestion
WI – contribution to the dose maxima from water ingestion
I-EE – contribution to the dose maxima from inhalation and external exposure

Table 7-6. *Annual landscape dose maxima to a representative person for other people, (E_{pop}), the year the maxima occur, the contribution to the dose maxima from different exposure pathways, and contributions from dominating radionuclides for selected cases (after table 7-8 in Hjerpe et al. 2010).*

Case	**E_{pop} [mSv]**	**Year**	**FI [%]**	**WI [%]**	**I/EE [%]**	**C-14**	**Cl-36**	**I-129**
Panel A								
Sh1	1.2×10^{-7}	3 970	33	67	0	84	2	14
Sh4 Q	4.7×10^{-6}	3 970	32	68	0	94	1	5
Panel B								
Sh1	2.0×10^{-8}	12 020	31	69	0	2	77	21
Sh4 Q	2.0×10^{-7}	3 570	0	100	0	0	1	99
Panel C								
Sh1	2.9×10^{-8}	3 570	0	100	0	100	0	0
Sh4 Q	4.5×10^{-7}	3 920	0	100	0	95	1	4

FI – contribution to the dose maxima from food ingestion
WI – contribution to the dose maxima from water ingestion
I-EE – contribution to the dose maxima from inhalation and external exposure



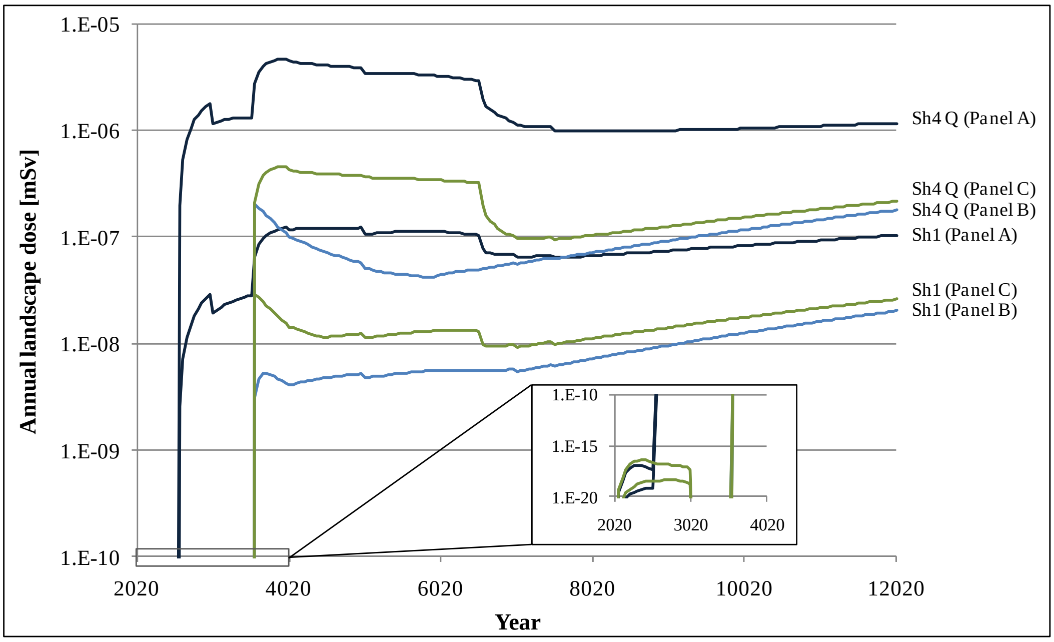


Figure 7-12. *Annual landscape doses to a representative person for the most exposed group, E_{group}, (top) and to a representative person for other people, E_{pop}, (below) for selected calculation cases (Figs 7-7 and 7-8 in Hjerpe et al. 2010).*

7.4.4 Typical absorbed dose rates to other biota

As described in Section 4.4, Guide YVL E.5 states that the disposal shall not affect other biota detrimentally, but no numerical constraints are given. The current Posiva methodology derives typical absorbed dose rates to assessment species (a group of species selected to cover different roles in the ecosystem) and compares the results with internationally proposed absorbed dose rate screening values for the protection of biota against radiation in the environment.

Derived typical absorbed dose rate maxima for the realistic biosphere calculation cases applied on the repository calculation cases Sh1 and Sh4 Q are presented in (Hjerpe et al. 2010). In these calculations, the dose rate ranges from about 6×10^{-6} to 2×10^{-3} μGy/h for terrestrial species and from about 2×10^{-9} to 4×10^{-3} μGy/h for freshwater/marine species. The terrestrial species with the highest dose rate maxima are, for both repository calculation cases, American mink, bank vole, common frog, hazel grouse, hooded crow, moose, mountain hare, red fox, tawny owl and viper. For species in freshwater, the highest dose rate maximum is estimated for grass snakes, and for marine species, the highest dose rate maximum is estimated for gray seals and oystercatchers.

7.5 Comparison with regulatory constraints

7.5.1 Single failed canister: the period up to several millennia after closure

The constraints on the annual dose for the most exposed individuals is 0.1 mSv, and the average annual dose for larger groups of people who may be exposed to radioactive releases is one hundredth to one tenth of the constraint for the most exposed individuals. These constraints apply for a period extending to a minimum over several millennia after the closure of the repository (see Section 4.4).

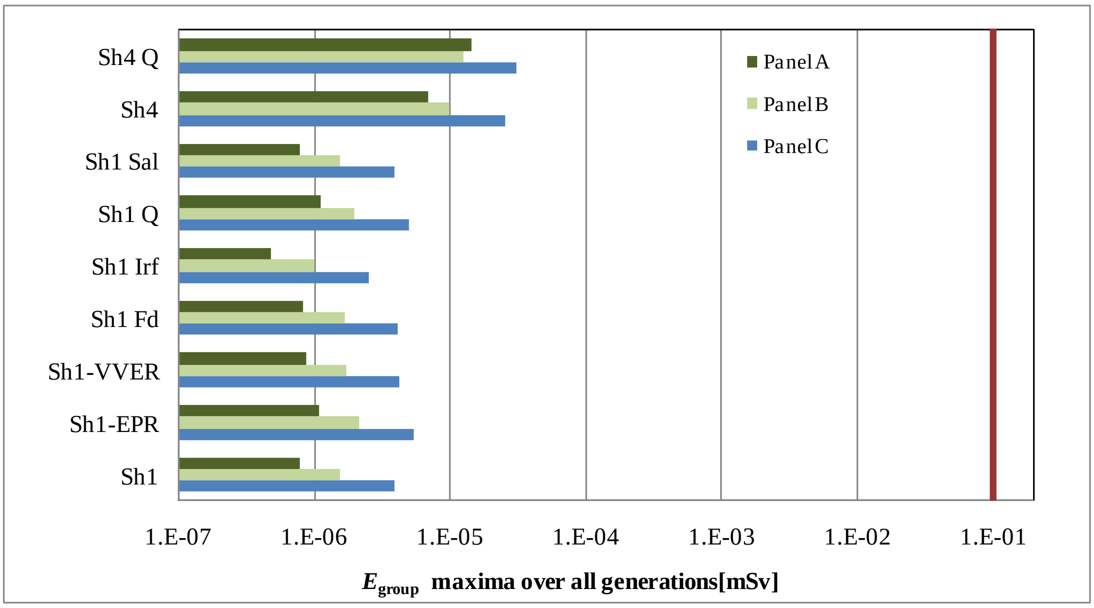
Typical absorbed dose rates to identified assessment species have been derived in the present assessment and are compared with internationally proposed screening values for the protection of biota against radiation in the environment and, in particular, the organism group-specific screening values recommended by the PROTECT project (Andersson et al. 2008). They are, in the form of absorbed dose rates, 2 μGy/h for vertebrates, 70 μGy/h for plants and 200 μGy/h for invertebrates.

Figure 7-13a shows the annual landscape dose maxima, over all generations, to the representative person for the most exposed group (E_{group}) for nine repository calculation cases. The highest doses are from case Sh4 Q, and are more than three orders of magnitude below the regulatory annual dose constraint.

Figure 7-13b shows the annual dose maxima, over all generations, to the representative person for the larger group (E_{pop}) for the same nine repository calculation cases. The highest doses are again from case Sh4 Q, and are more than two orders of magnitude below the lower limit of the regulatory annual dose constraint band.

Figure 7-14a to Figure 7-14c show the highest estimated typical absorbed dose rates for each identified assessment species. The results show that the assessed typical absorbed dose rates are more than two orders of magnitude below the lowest proposed screening value.

(a)



(b)

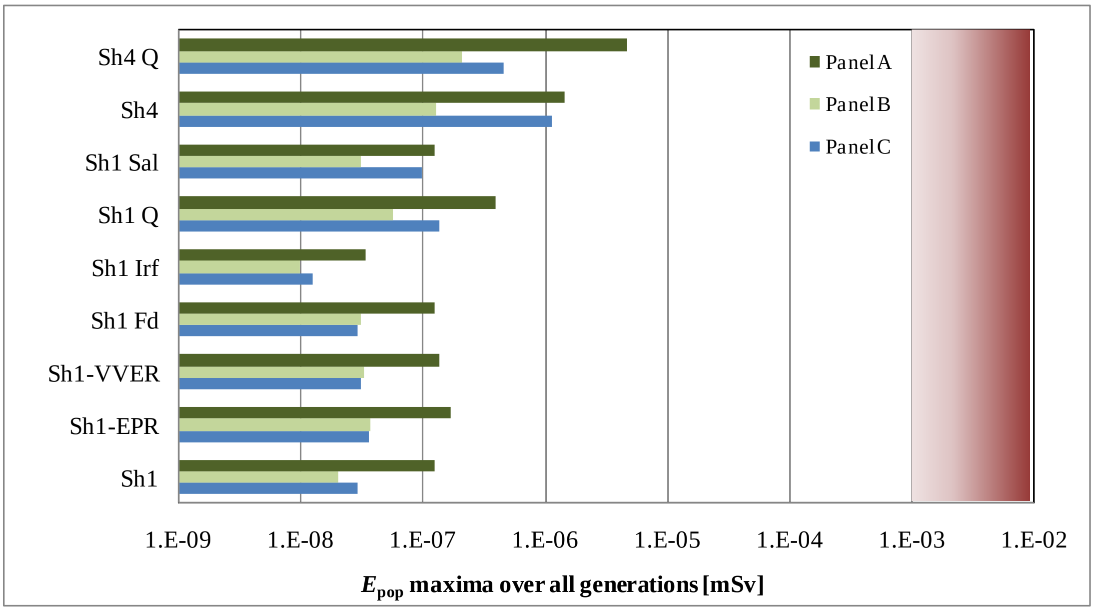
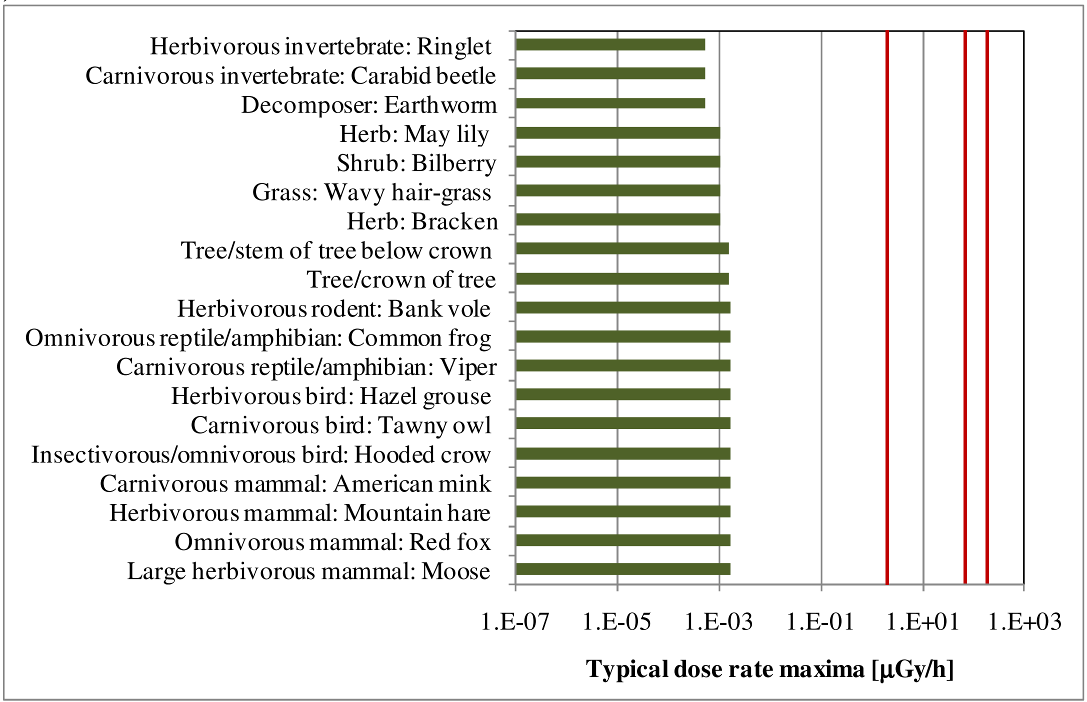
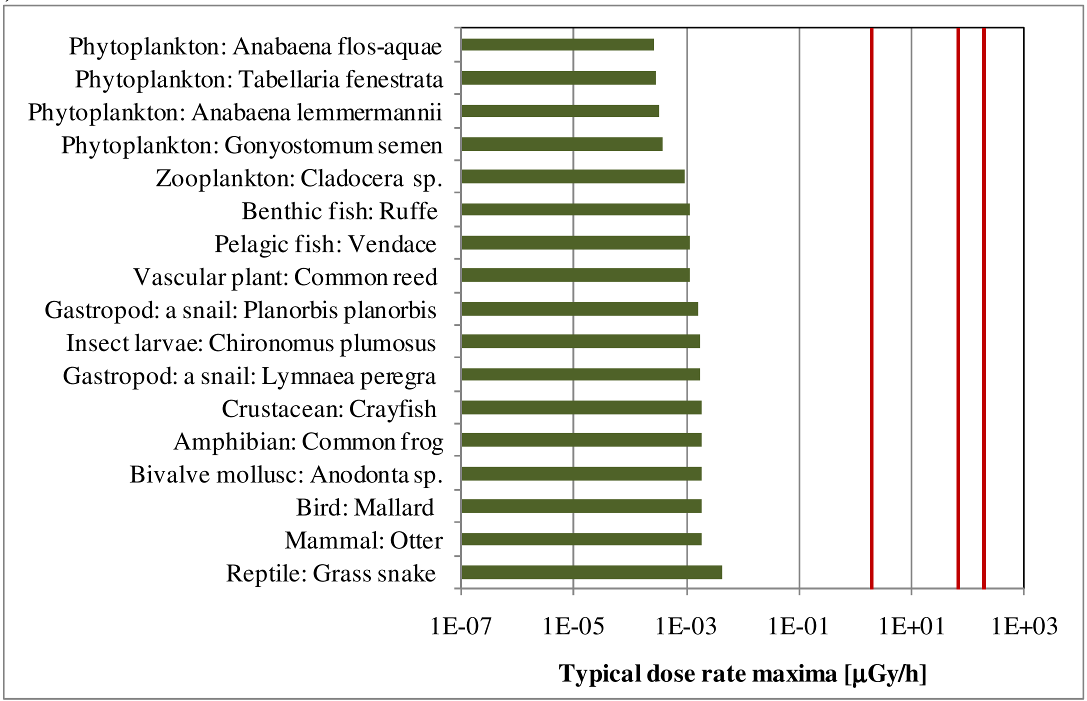


Figure 7-13. *Annual landscape dose maxima, over all generations, (a), to the representative person for the most exposed group (E_{group}) and (b), to the representative person for other people (E_{pop}), for nine repository calculation cases. The red line in the upper figure corresponds to the regulatory annual dose constraint to the most exposed people. The graded red box in the lower figure corresponds to the regulatory average annual dose constraint band to other people (Figs. 10-1 and 10-2 in Hjerpe et al. 2010).*

(a)


Herbivorous invertebrate: Ringlet
Carnivorous invertebrate: Carabid beetle
Decomposer: Earthworm
Herb: May lily
Shrub: Bilberry
Grass: Wavy hair-grass
Herb: Bracken
Tree/stem of tree below crown
Tree/crown of tree
Herbivorous rodent: Bank vole
Omnivorous reptile/amphibian: Common frog
Carnivorous reptile/amphibian: Viper
Herbivorous bird: Hazel grouse
Carnivorous bird: Tawny owl
Insectivorous/omnivorous bird: Hooded crow
Carnivorous mammal: American mink
Herbivorous mammal: Mountain hare
Omnivorous mammal: Red fox
Large herbivorous mammal: Moose
1.E-07
1.E-05
1.E-03
1.E-01
1.E+01
1.E+03
Typical dose rate maxima [µGy/h]

(b)


Phytoplankton: Anabaena flos-aquae
Phytoplankton: Tabellaria fenestrata
Phytoplankton: Anabaena lemmermannii
Phytoplankton: Gonyostomum semen
Zooplankton: Cladocera sp.
Benthic fish: Ruffe
Pelagic fish: Vendace
Vascular plant: Common reed
Gastropod: a snail: Planorbis planorbis
Insect larvae: Chironomus plumosus
Gastropod: a snail: Lymnaea peregra
Crustacean: Crayfish
Amphibian: Common frog
Bivalve mollusc: Anodonta sp.
Bird: Mallard
Mammal: Otter
Reptile: Grass snake
1E-07
1E-05
1E-03
1E-01
1E+01
1E+03
Typical dose rate maxima [µGy/h]

(c)

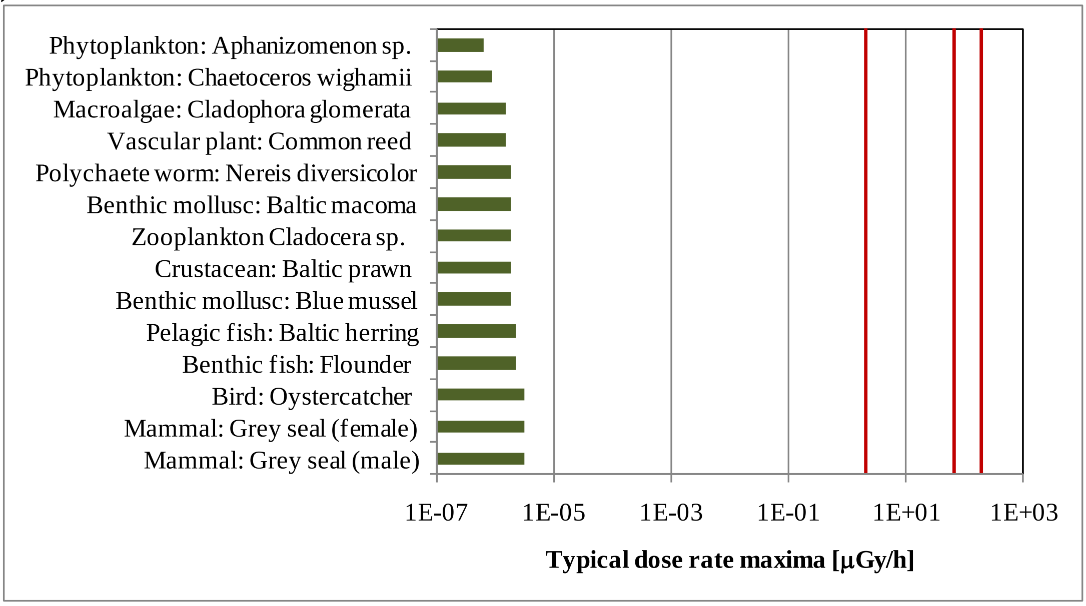


***Figure 7-14**. Typical absorbed dose rate maxima for (a), terrestrial assessment species, (b), assessment species in freshwater and (c), assessment species in water of the Baltic coast. All maxima occur for the repository base case Sh4 Q with releases from either (a), repository panel C, or (b) and (c), repository panel A. The red lines correspond to the applied screening values (2 µGy/h for vertebrates, 70 µGy/h for plants and 200 µGy/h for invertebrates) (Figs. 10-3 to 10-5 in Hjerpe et al. 2010).*

7.5.2 Single failed canister: the period after several millennia

In the longer term, after several millennia, the quantitative regulatory criteria are based on constraints, given in Guide YVL E.5, on the release rates of long-lived radionuclides from the geosphere to the biosphere:

"*The sum of the ratios between the nuclide-specific activity releases and the respective constraints shall be less than one*".

Figure 7-10 shows the maxima of the calculated overall release rate ratios and their times of occurrence for the various calculation cases.

The highest value of release rate ratio occurs in case RS1 of the rock shear/earthquake scenario. In this least favourable case, the maximum release rate ratio is more than an order of magnitude below the regulatory constraint.

7.5.3 Likelihood or rate of canister failure

The safety analysis described in the previous sections considers the consequences of the failure of a single canister by different potential failure modes. Estimating the likelihood or rate of canister failure is an objective of ongoing work (see Ch. 11).

So far, tentative estimates have been made of the number of canisters that might fail in the event of a large earthquake occurring in the vicinity of the repository, since the information is already available to make such estimates for this particular canister failure mode (it is also the canister failure mode giving the highest consequences in the case of a single canister failure, as shown in Fig. 7-15). La Pointe and Hermansson (2002) studied fracture displacements with respect to potential future seismicity for Olkiluoto, considering earthquakes of magnitudes (M_L) of 5.5 through 7...8, and estimated that 6 canisters out of 3 000 could fail in the event of a large earthquake and that the probability of such an earthquake is 0.02 over a 100 000 year period. More recently, the expectation value of the number of canisters in the repository that could potentially be damaged in the event of a large earthquake has been estimated to be 16 out of the total number of 3 000 canisters for a KBS-3H repository, and 20 for a KBS-3V repository (see the KBS-3H Evolution Report, Smith et al. 2007b, p. 145). Multiple earthquakes would not necessarily increase this estimate, since it represents the total number of vulnerable canister positions – i.e. drift sections intersected by large fractures. Furthermore, no rock suitability criteria were applied in deriving this estimate (since none have so far been developed for a KBS-3H repository). Thus, the estimate of 16 to 20 failed canisters represents a pessimistic upper bound for this failure mode.

Based on the results given in Figure 7-15, even assuming that all 20 canisters in vulnerable positions fail, and do so simultaneously, the Finnish regulatory geo-bio flux constraint are still met.

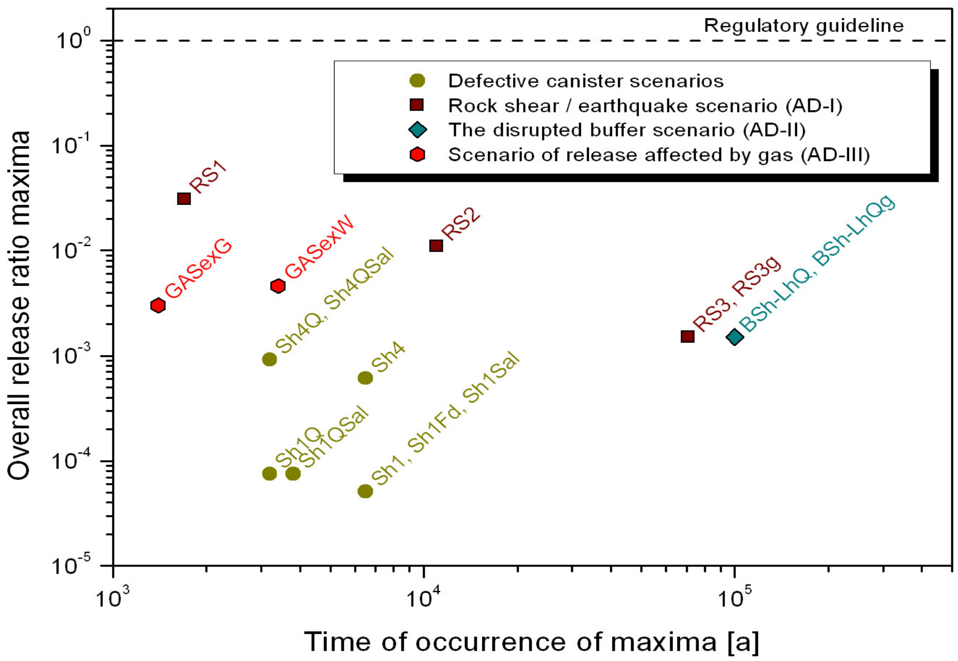


Figure 7-15. *The maxima of overall release rate ratio (as defined in the main text) and their times of occurrence in a selection of calculation cases for each scenario. Note that these cases represent a subset of those assessed in the 2009 KBS-3V safety analysis.*

8 ANALYSIS OF SCENARIOS IN THE KBS-3H SAFETY ANALYSIS

This chapter (Fig. 8-1) presents a summary of the results of the analysis of scenarios in the KBS-3H safety assessment.

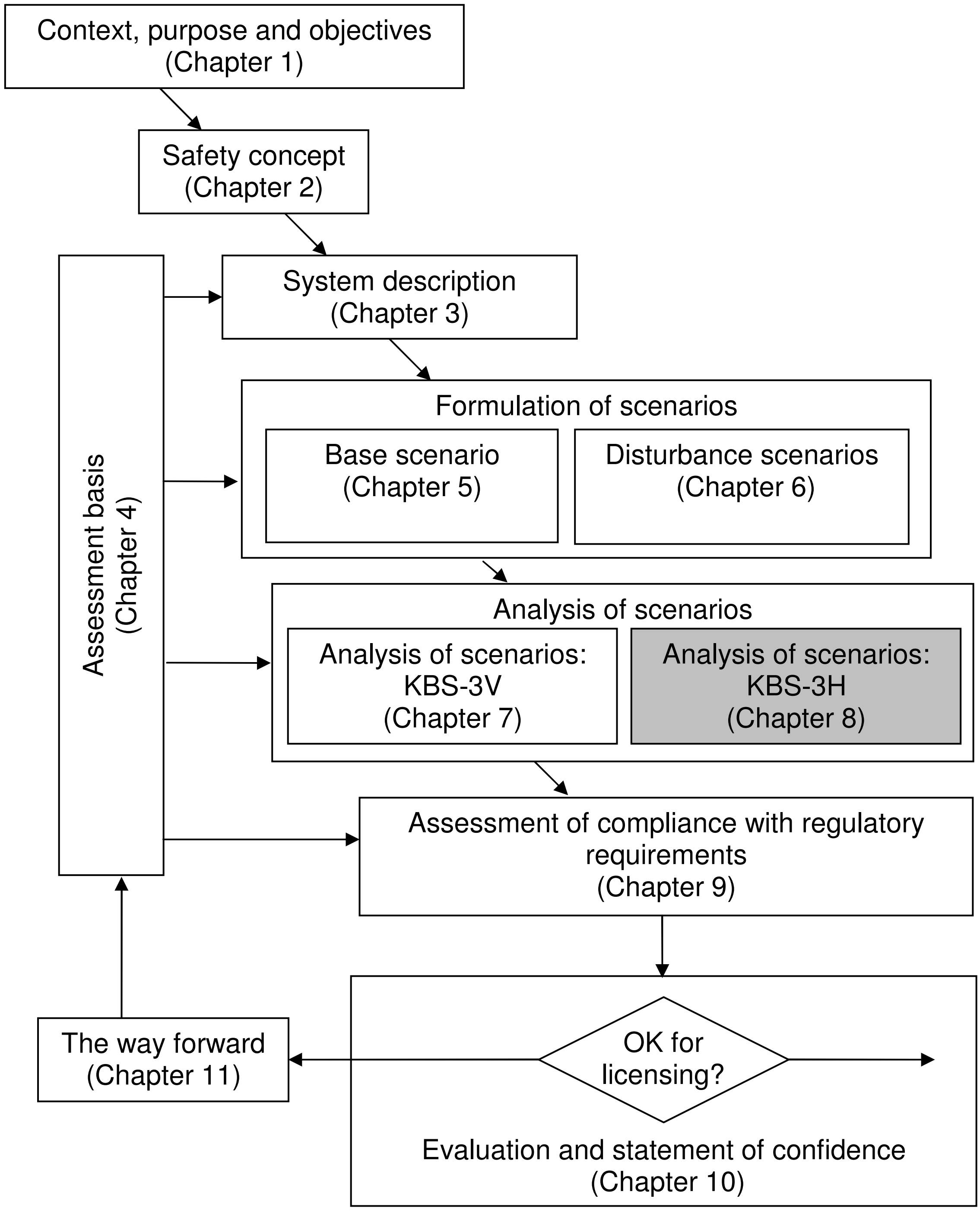


Figure 8-1. *The present chapter in the context of the safety case summary report.*

The chapter is structured as follows:

- Section 8.1 describes the scenarios and calculation cases considered in the safety analysis;
- Section 8.2 presents the releases to the biosphere evaluated for a canister with an initial penetrating defect;
- Section 8.3 addresses other canister failure modes;
- Section 8.4 comments on the differences in calculated radionuclide release rates for KBS-3H and KBS-3V;
- Section 8.5 presents the evaluation of landscape dose; and
- Section 8.6 compares the results of the analyses with regulatory constraints.

The chapter summarises work presented in full in Smith et al. (2007c) and in Broed et al. (2007). In future, the analysis of scenarios will be described in the Analysis of Scenarios Report (Fig. 1-3).

8.1 Organisation of the safety analysis

The focus of the KBS-3H safety assessment was on uncertainties that are specific to the KBS-3H variant, or have different implications for KBS-3H compared with KBS-3V. By contrast, the safety analysis of a KBS-3V repository (Nykyri et al. 2008) considered a wider range of uncertainties, and included calculation cases that demonstrated the robustness of the repository design, illustrated the relative importance of the components of the multi-barrier system, examined the sensitivity to variations of key parameters, and also explored the system's behaviour in extreme situations.

The repository calculation cases considered in the KBS-3H safety assessment were also organised somewhat differently from those in the safety analysis of a KBS-3V repository (Nykyri et al. 2008). In particular, the assessment cases were grouped according to the canister failure mode that they addressed (in the terminology of the KBS-3H safety assessment, canister failure by a given mode may result from one or more scenarios). Each case was assigned a unique name, comprising two parts separated by a hyphen. The first part of the name indicates the canister failure mode that the case addresses:

- PD: canister with an initial penetrating defect;
- CC: canister failure due to copper corrosion; and
- RS: canister failure due to rock shear.

The second part of the name identifies the case either as a base calculation case (BC) for a given canister failure mode, or a variant case illustrating the impact of one or more uncertainties. Thus, PD-BC is the base calculation case for the group of calculation cases addressing a canister with an initial penetrating defect. As in the 2009 safety analysis of a KBS-3V repository, the majority of calculation cases were defined for a scenario in which a canister has an initial penetrating defect.

The biosphere analysis was more limited than that in the 2009 safety analysis of a KBS-3V repository, in the sense that, in the KBS-3H safety assessment, only a single model representation of the biosphere was used for the evaluation of dose due to the release calculated for those repository calculation cases where radionuclide release begins within the first ten thousand years. The results of the biosphere analysis performed for the KBS-3H safety assessment are summarised in Section 8.4.

8.2 Canister with an initial penetrating defect

8.2.1 The base calculation case

The canister failure due to an initial penetrating defect, termed the PD failure mode, corresponds to scenario DCS-II in the 2009 safety analysis of a KBS-3V repository. In the base calculation case (PD-BC), as in the corresponding KBS-3V base calculation case, the diameter of the initial penetrating defect was taken to be 1 mm (a larger, 4 mm defect was also considered in the variant cases). The modelling of a canister with an initial penetrating defect was, however, carried out somewhat differently compared with the KBS-3V safety analysis In particular, in the KBS-3H safety assessment, it was assumed to take 1 000 years for water to contact the fuel, the Zircaloy and other metal parts and for a transport pathway to become established between the canister interior and exterior, following a similar assumptions in the SR-Can safety assessment (Section 10.5.2 of SKB 2006a). These events were conservatively assumed to occur instantaneously upon canister emplacement in the KBS-3V safety analysis.

After 1 000 years, in case PD-BC the defect provides a continuing transport resistance for released radionuclides for a further 9 000 years, before final loss of this transport resistance at 10 000 years after canister deposition. It should be noted that the SR-Can Data Report (Section 4.4.7 of SKB 2006c) suggests that loss of transport resistance may occur at any time between 1 000 and 100 000 years after radionuclide transport pathways from the canister interior are established, and the choice of 10 000 years as the PD-BC parameter value was somewhat arbitrary. Furthermore, loss of transport resistance may be a process that occurs gradually over time, rather than as a discrete event. An instantaneous loss of transport resistance is, however, a conservative assumption, since a gradual loss of transport resistance will spread the peak release over a longer period of time, reducing the maximum rate of release.

The geometry of the domain represented by the PD-BC near-field model is the same as that shown in Figure 4-5. As in the 2009 safety analysis of a KBS-3V repository, the defect is pessimistically located at a position that minimises the transport distance across the buffer between the defect and the fracture mouth.

Figure 8-2 shows the release ratio for calculation case PD-BC as a function of time. The term “release ratio” is defined in Section 7.2.1. The figure shows the overall (total) release ratio and the contributions from the most important individual radionuclides. There is a sharp increase in the release ratio starting at about 9 000 years, associated with the loss of transport resistance of the defect. As noted above, loss of transport resistance is taken to occur at 10 000 years in the near-field model. The earlier calculated onset of the dose increase is an artefact of the finite modelling time step. The

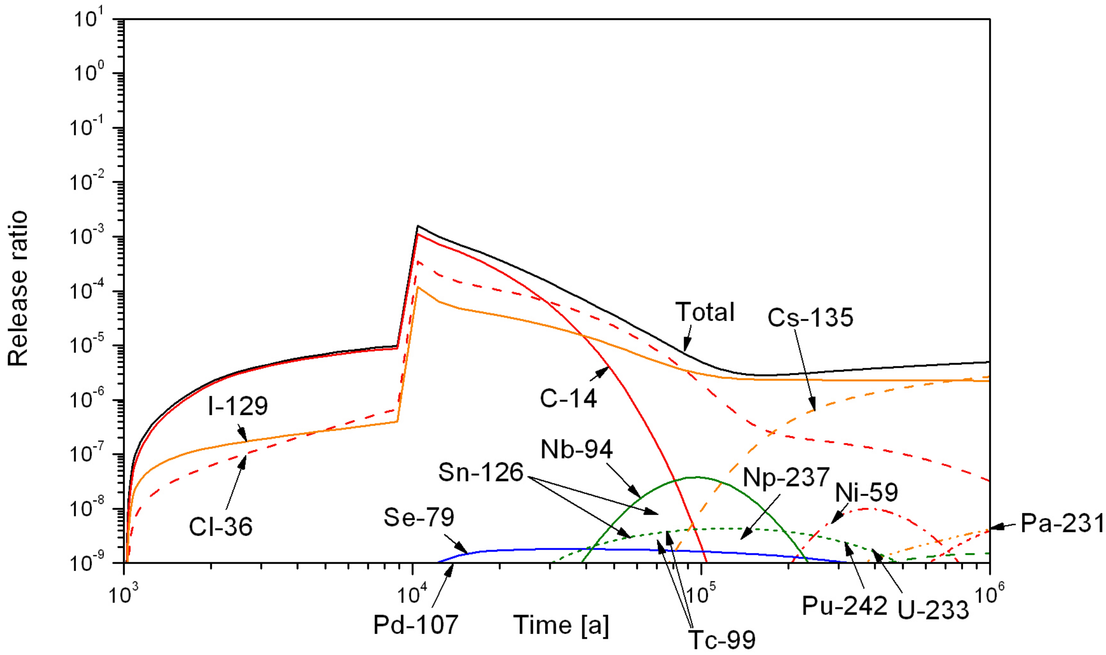


Figure 8-2. *Release ratio as a function of time in case PD-BC.*

overall release ratio maximum (2×10^{-3} for this case for a single canister failure) occurs shortly after loss of transport resistance of the defect. The overall release ratio is dominated by C-14 up to about 20 000 years, by Cl-36 up to about 90 000 years, and thereafter by I-129 and, beyond a few hundred thousand years, by Cs-135.

As in the KBS-3V safety analysis, it is radionuclides such as I-129, Cl-36 and C-14 that generally dominate calculated releases and doses until near the end of the assessment period, since these are long-lived and undergo little or no sorption in the buffer and geosphere. Although not calculated, Cs-135 release to the biosphere is expected to fall shortly after a million years as its inventory becomes depleted by radionuclide decay, its half life being 2.3×10^{6} years.

8.2.2 Impact of perturbations to the buffer/rock interface

There are a number of features and processes identified in 6.2.1 that could significantly affect the properties of the buffer/rock interface. In the base calculation case, it was assumed that the impact of these processes on radionuclide release and transport is negligible. The degree to which system properties will, in reality, be affected and the spatial extent of any perturbation is, however, highly uncertain. Thus, four variant cases were considered in which the impact of these features and processes was assumed to be more significant. All four cases are similar, in that the perturbing processes are taken to create a high-permeability zone at the buffer/rock interface. The extent of the zone, and the groundwater flow within the zone, are, however, case dependent.

Case PD-SPALL addresses thermally-induced rock spalling in a relatively tight drift section, where buffer swelling pressure at the drift wall is not developed rapidly enough to prevent this process from taking place. It could also be considered to address the impact of porous iron corrosion products being in direct contact with the drift wall – this is also a possibility in relatively tight drift sections. Although the hydraulic conductivity of the bedrock near the drift wall in affected drift sections is assumed to be substantially increased by thermally induced spalling, it is also assumed that the pressure on the drift wall exerted by the distance blocks suppresses spalling and prevents the formation of continuous flow and transport pathways along the drift.

Cases PD-FEBENT1, PD-FEBENT2 and PD-FEBENT3 address chemical interactions of the buffer with the iron of the supercontainer (or with high-pH leachates from cementitious repository components). In each case, the zone affected by these interactions is treated as a "mixing tank". Radionuclides entering the zone become uniformly mixed with the groundwater flowing through the zone. Groundwater flow in the zone is calculated on the assumption that the hydraulic conductivity is much higher than that of either the rock or the buffer (under this pessimistic assumption, the flow becomes independent of the exact value assigned to the zone hydraulic conductivity). In case PD-FEBENT1, the extent of the affected buffer zone is limited to a region around the supercontainer of vanishingly small thickness, but this region is assigned a high (effectively infinite) hydraulic conductivity. In case PD-FEBENT2, the affected buffer zone is assumed to extend across 10% (4 cm) of the entire thickness of the buffer. This is consistent, for example, with the results of reactive transport modelling (Wersin et al. 2007). In case PD-FEBENT3, the affected buffer zone is conservatively assumed to extend across half the entire thickness of the buffer (20 cm).

In Figure 8-3, the evolution of the overall release ratios for all four cases addressing perturbations to the buffer/rock interface are compared with the base calculation case for a canister with an initial penetrating defect (PD-BC). Releases in case PD-SPALL are slightly reduced with respect to the base calculation case, since, although the buffer/rock interface is unfavourably perturbed, thermally-induced rock spalling is assumed to affect only a relatively tight drift section. The release maxima in cases PD-FEBENT1, PD-FEBENT2 and PD-FEBENT3 are similar to each other (the thickness of the perturbed buffer/rock interface zone is an insensitive parameter), and increased by about an order of magnitude with respect to the base calculation case.

8.2.3 Expulsion of contaminated waters from the canister interior by gas

The possibility of expulsion of contaminated water from the canister interior by gas is considered in the KBS-3H calculation case PD-EXPELL, which is analogous to case GASexW of the AD-III scenario in KBS-3V safety analysis, and is analysed using a similar model and parameter values (note that calculation cases analogous the KBS-3V case GASexG of the AD-III scenario were also considered, see Smith et al. 2007c). The calculated overall and nuclide-specific release ratios are shown in Figure 8-4. The maximum, which occurs shortly after the start of water expulsion, is more than an order of magnitude higher than that of the base calculation case (PD-BC).

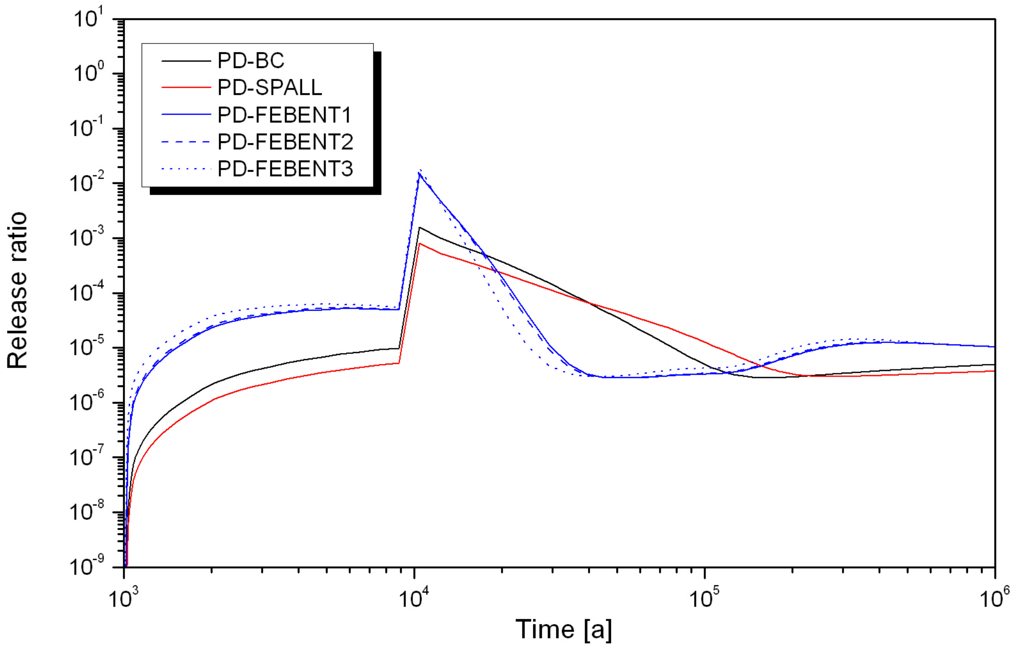


Figure 8-3. *Overall release ratios in all four calculation cases addressing perturbations to the buffer / rock interface and in the base calculation case (PD-BC).*

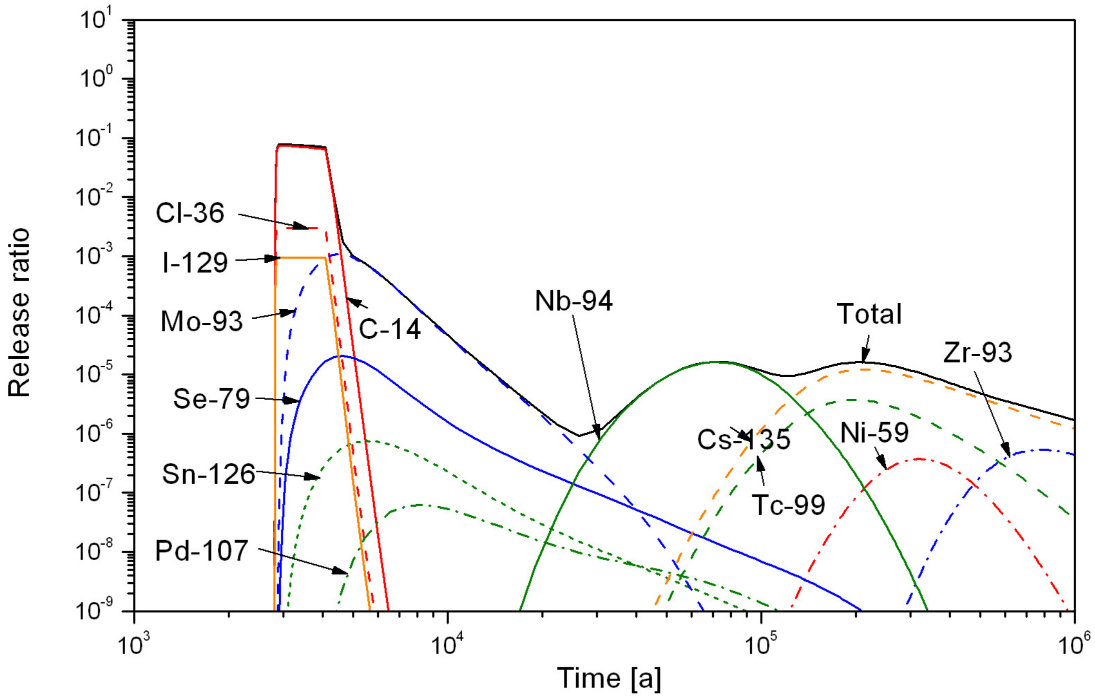


Figure 8-4. *Nuclide-specific and overall release ratios in calculation case PD-EXPELL.*

8.3 Other canister failure modes

8.3.1 Failure due to copper corrosion

Canister failure due to copper corrosion, termed the CC failure mode, corresponds to scenario AD-II in the KBS-3V safety analysis, and the same model assumptions are applied in the analysis of calculation cases, with the exception that the canister contains no initial penetrating defect, and thus there is no radionuclide release prior to canister failure, which occurs at 100 000 years in the future, following buffer erosion due to an influx of glacial meltwater[15]. Figure 8-5 shows the nuclide-specific and overall release ratios as functions of time in the base calculation case for this canister failure mode. The maximum, which is dominated by I-129 followed by Cl-36, occurs shortly after canister failure at 100 000 years. At later times, the overall release ratio is dominated firstly by Cl-36, and later by Se-79, Cs-135 and finally, after about 500 000 years, by Th-229 (a decay product of Np-237).

It should be noted that the magnitude of the peak shortly after 100 000 years is increased by the conservative assumption that the canister fails completely and instantaneously at this time. In reality, some mass transport resistance may be retained following the initial failure of a canister.

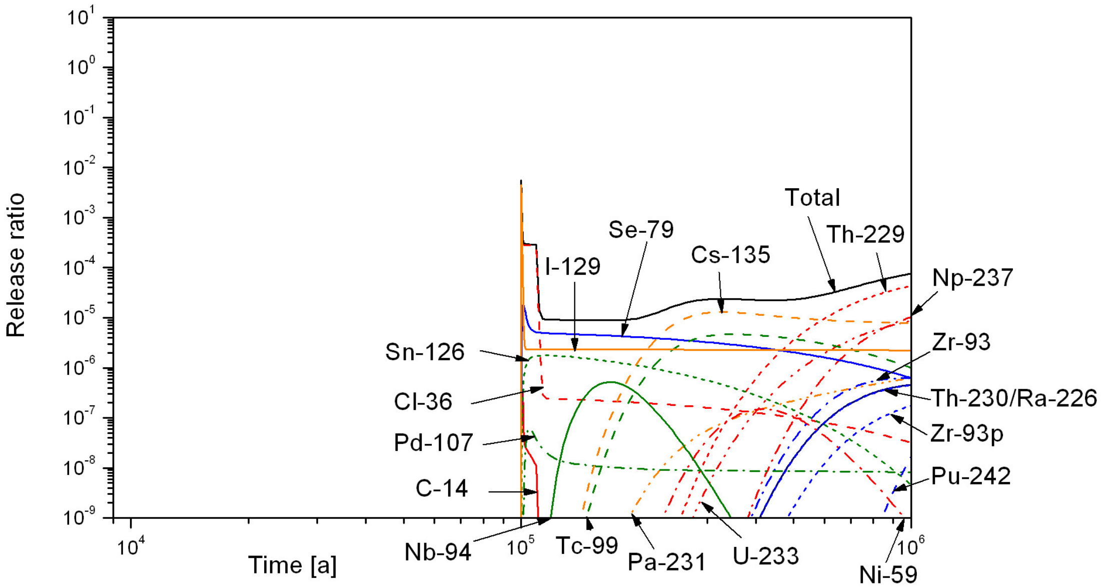


Figure 8-5. *Nuclide-specific and overall release ratios as functions of time in calculation case CC-BC.*

[15] As noted in Section 7.3.1, 70 000 years in the future would be the time of the next glacial retreat, assuming a repetition of the last glacial cycle. Considerable further time will, however, be required for canister failure by corrosion, even if advective conditions become established in the buffer (Smith et al. 2007c).

8.3.2 Failure due to rock shear

Canister failure due to rock shear, termed the RS failure mode, corresponds to scenario AD-I in the KBS-3V safety analysis (Nykyri et al. 2008), and similar model assumptions are applied in the analysis of calculation cases, although only the possibility of canister failure at 70 000 years was considered (as in the KBS-3V calculation cases RS3 and RS3g). Figure 8-6 shows the nuclide-specific and overall release ratios as functions of time in the base calculation case for this canister failure mode (RS-BC). The maximum occurs shortly after canister failure at 70 000 years. The highest overall release ratio, however, occurs at later times, (5×10^{-3} at a million years) and is due to Ra-226 (a decay product of Th-230, U-234 and U-238).

A reduced geosphere transport resistance compared with PD-BC takes into account the possible detrimental impact that rock shear may have on the geosphere transport barrier. This means that there is only limited attenuation of near-field releases of many radionuclides by decay during geosphere transport in case RS-BC (see Section 7.2.4 of the Smith et al. 2007c). In particular, Ra-226 and its parent radionuclide Th-230 are significantly less attenuated by decay during geosphere transport than in the base calculation case for an initial penetrating defect (PD-BC), explaining the much higher long-term release rate of Ra-226 to the biosphere compared with case PD-BC.

8.4 Comments on differences between KBS-3H and KBS-3V releases

Generally, a comparison of results from the 2009 safety analysis of a KBS-3V repository and from the KBS-3H safety assessment is not a meaningful exercise, given the different model assumptions and parameter values selected for the two safety assessments. For example, in the case of an initial penetrating defect, the 2009 safety analysis of a KBS-3V repository assumed immediate ingress of water to the canister interior and further that the defect remains unchanged for all time. The KBS-3H safety assessment, on the other hand, assumed that ingress of water requires 1 000 years to occur and that the transport resistance of the defect is lost 10 000 years after canister deposition.

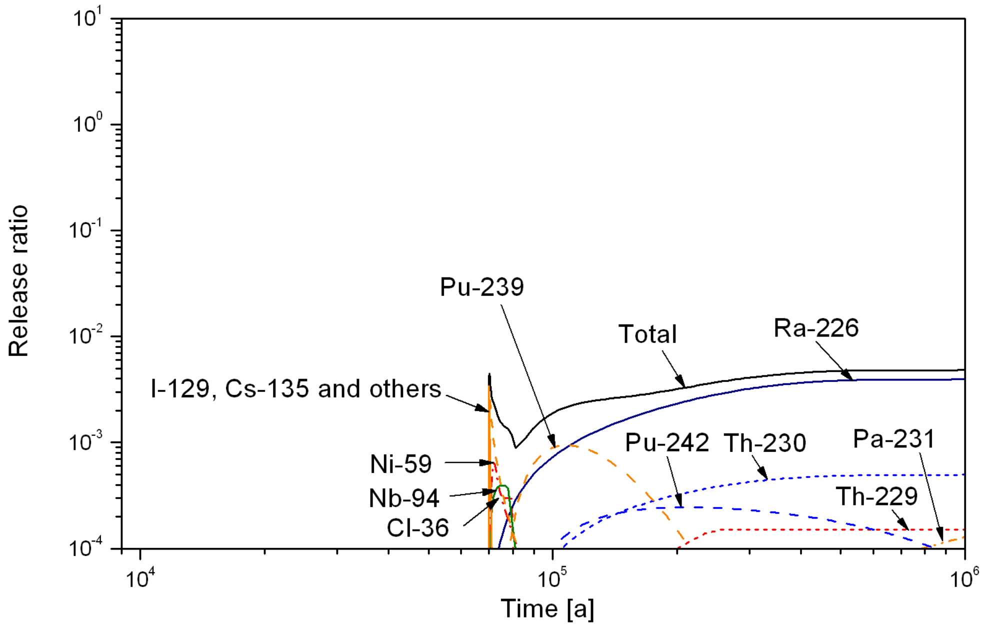


Figure 8-6. *Ratios of nuclide-specific activity releases to their respective geo-bio flux constraints in calculation case RS-BC.*

To facilitate a meaningful comparison, the 2009 safety analysis of a KBS-3V repository also included a calculation case for a KBS-3V repository that replicates the assumptions from the KBS-3H base calculation case for an initial penetrating defect (PD-BC). The KBS-3V case is named "Sh1 as PD-BC", and is shown in Figure 8-7, along with PD-BC itself. The results indicate that differences in the geometry and transport paths considered in the analysis of the KBS-3V and KBS-3H design variants have only a minor impact on calculated releases and doses.

8.5 Dose assessment

The biosphere analysis performed in the KBS-3H safety assessment is documented in Broed et al. (2007), and in several supporting reports (e.g., Broed 2007, Ikonen 2007), including the conceptual and mathematical models and key data used in landscape modelling and the radiological consequence analysis. The scope of the biosphere analysis was limited to derivation of annual doses to the most exposed people. It should be stressed that the landscape dose estimates in the KBS-3H safety analysis are considered to be pessimistic. The main reason is the level of conservatism implemented throughout the assessment process. In the KBS-3H biosphere analysis, there are components in the modelling process for estimating landscape doses that, knowingly, have an inherent excess of conservative assumptions. This concerns the conceptualisation of radionuclide transport models and the selection of parameter values, as well as the approach to identifying doses for use in the compliance assessment. The reason for this excessive conservatism is that the whole landscape dose concept was at a rather early stage of development when conducting the KBS-3H analysis. The level of conservatism underlying the dose estimates in the KBS-3V biosphere assessment (Section 7.4) is considered to be more adequate. It should be noted that the differences in the level of conservatism in the KBS-3V and KBS-3H biosphere analyses do not affect the comparison of doses shown in Fig. 8-7, since the comparison is limited to well doses.

Dose assessments are explicitly required by regulations only in cases where there are calculated releases to the biosphere within the time window from emplacement up to several thousand years in the future. It was found that calculated releases occur in the time frame only where an initial penetrating defect is present in a canister (and only in a subset of the calculation cases defined for this canister failure mode). It is for these cases that annual landscape doses have been estimated.

- PD-BC: the base calculation case of the initial penetrating defect canister failure mode;
- PD-LODELAY: a variant considering a reduced delay until loss of transport resistance of the defect;
- PD-LOGEOR: a variant with reduced geosphere transport resistance;
- PD-HISAL: a variant with saline water present at repository depth at all times;
- PD-EXPELL: a variant that considers dissolved radionuclides expelled by gas from the canister interior and across the buffer to the geosphere;
- PD-FEBENT3: a variant considering a perturbed buffer / rock interface, where the pessimistic assumption is made that the perturbed zone extends into the buffer over half its thickness; and
- PD-VOL-1: a variant with transportation of C-14 as volatile species by gas;

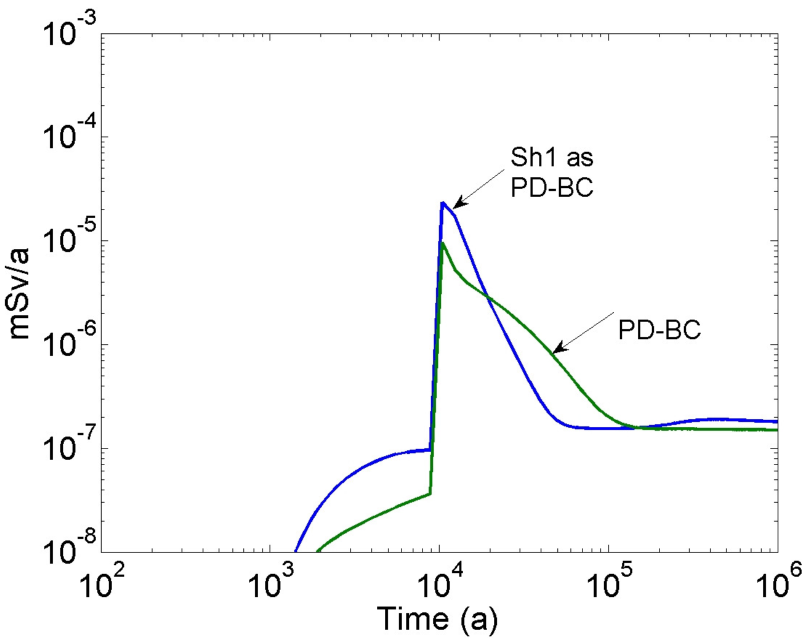


***Figure 8-7**. Calculated dose rates as a function of time in the case of a canister with an initial penetrating defect. Curve "Sh1 as PD-BC" shows the WELL-2008 dose from the safety analysis of the KBS-3V concept (Nykyri et al. 2008). Curve "PD-BC" shows the WELL-2007 dose from the KBS-3H Radionuclide Transport Report (Smith et al. 2007c). WELL-2007 dose is defined in Section 4.4.*

The calculated nuclide-specific and total annual landscape doses to the most exposed individual in the seven representative assessment cases are shown in Figures 8-8 to 8-11 (the nuclide-specific landscape dose for PD-VOL-1 Is the same as the total, since the release only contains C-14). Other cases assuming an initial penetrating defect have been treated by scaling approaches or qualitative arguments, as described in Broed et al. 2007. Other canister failure modes occur after the "environmentally predictable future" and so no evaluation of annual doses is required.

8.6 Comparison with regulatory constraints

8.6.1 Single failed canister: the period up to several thousand years after closure

As illustrated in Figure 8-11, of the six repository calculation cases for which annual landscape dose has been calculated, the highest annual effective dose occurs to the most exposed *individual* in case PD-VOL-1, with a maximum of about 6 × 10^{-2} mSv occurring at about 1 900 years. This is a factor of 2 below the regulatory constraint of 0.10 mSv for the most exposed people. For all other assessment cases, the maxima range from 5 × 10^{-4} to 2 × 10^{-2} mSv, and are thus around one to two orders of magnitude below the regulatory constraint, even though these annual doses as calculated are likely to be more conservative quantities than required for the comparison. It should, however, again be noted that only a single failed canister is considered in each case.

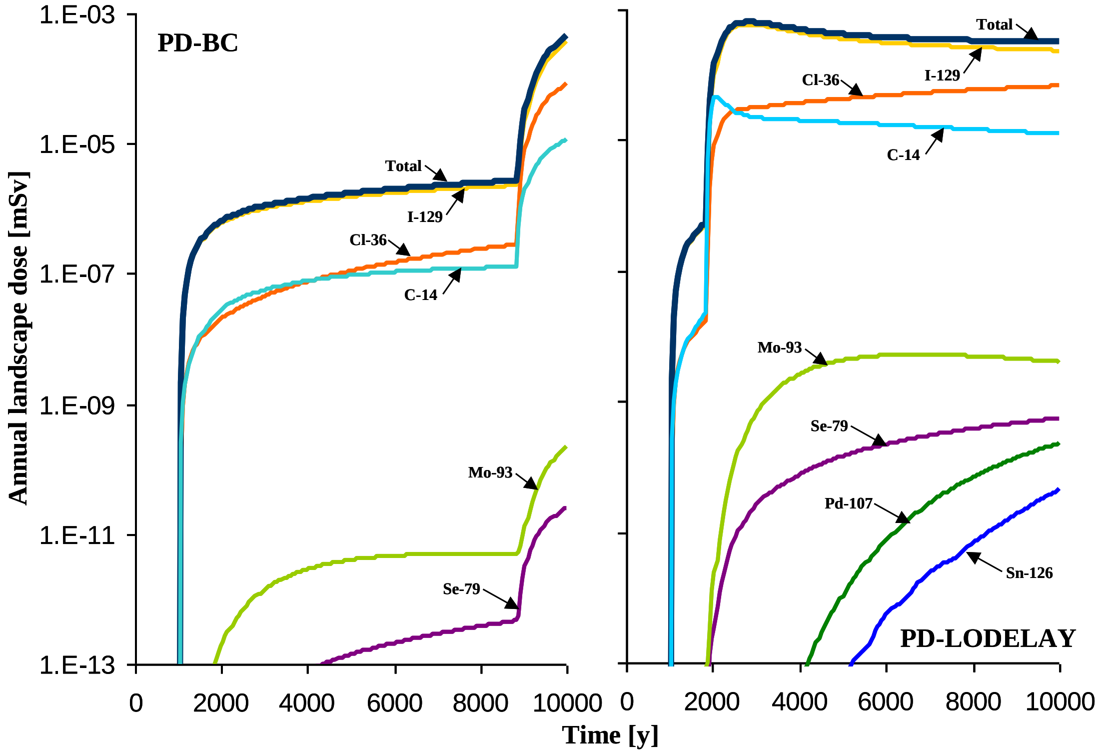


Figure 8-8. *Annual landscape doses during the first 10 000 years arising from the calculation cases PD-BC and PD-LODELAY (Fig. E-1 in Broed et al. 2007).*

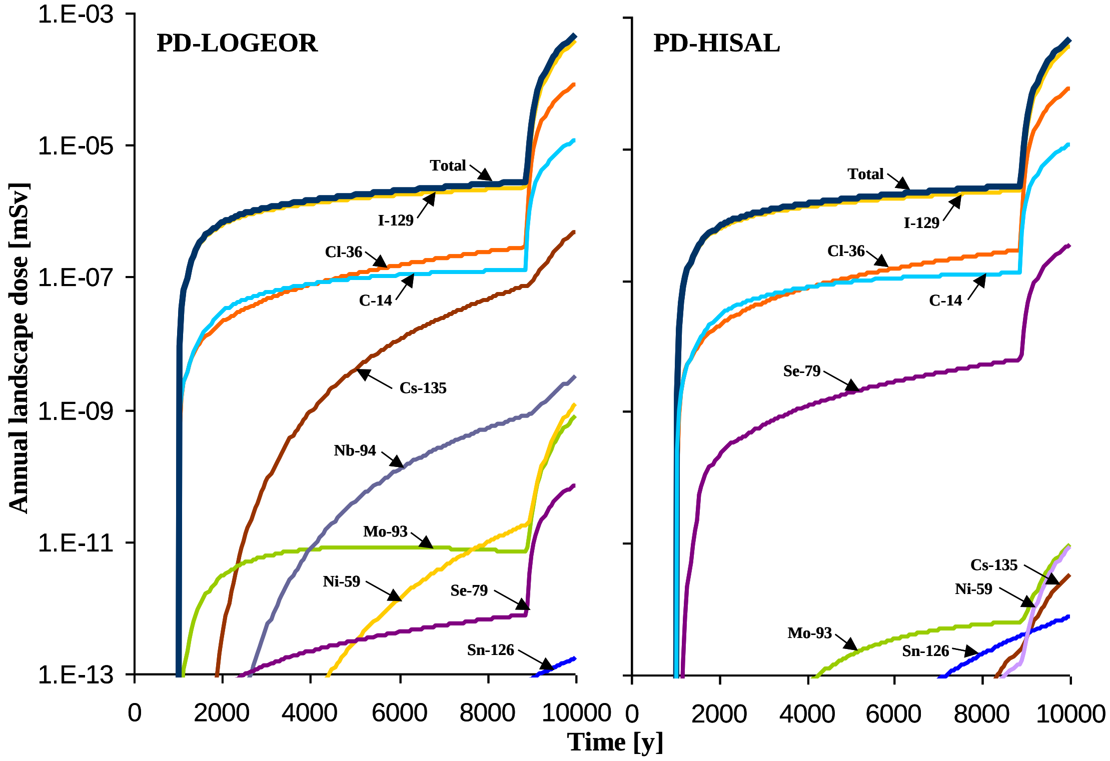


Figure 8-9. *Annual landscape doses during the first 10 000 years arising from the calculation cases PD-LOGEOR and PD-HISAL (Fig. E-2 in Broed et al. 2007).*

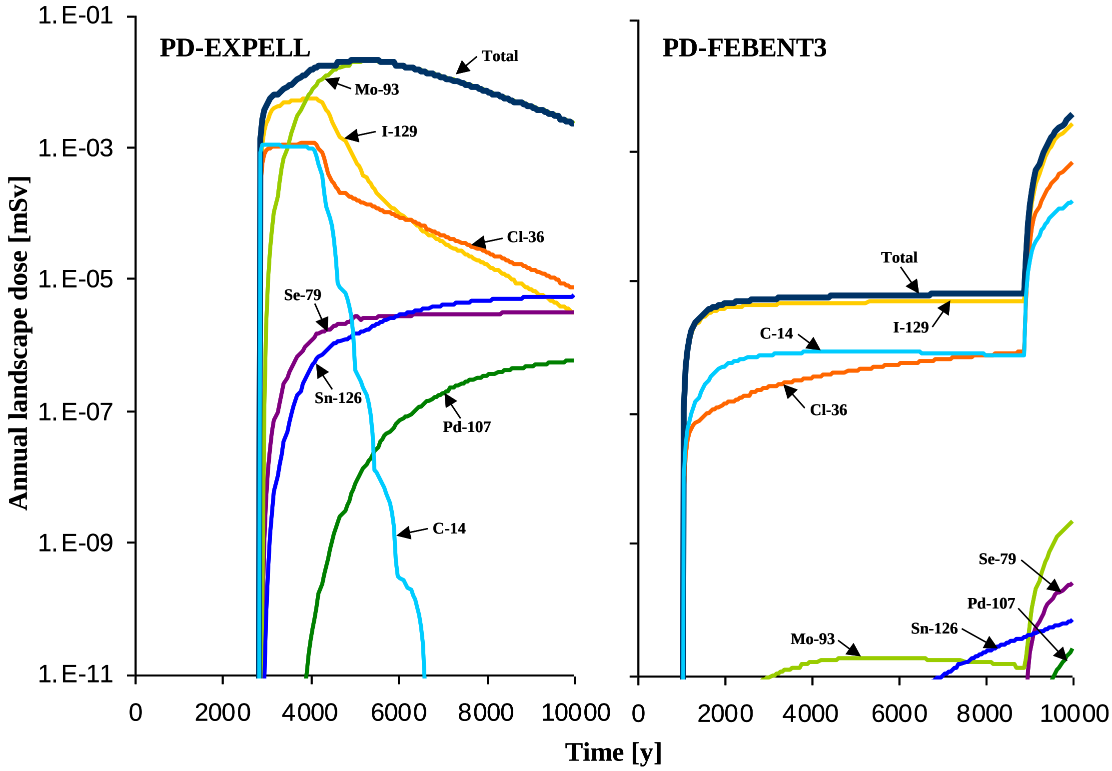


Figure 8-10. *Annual landscape doses during the first 10 000 years arising from the calculation cases PD-EXPELL and PD-FEBENT3 (Fig. E-3 in Broed et al. 2007).*

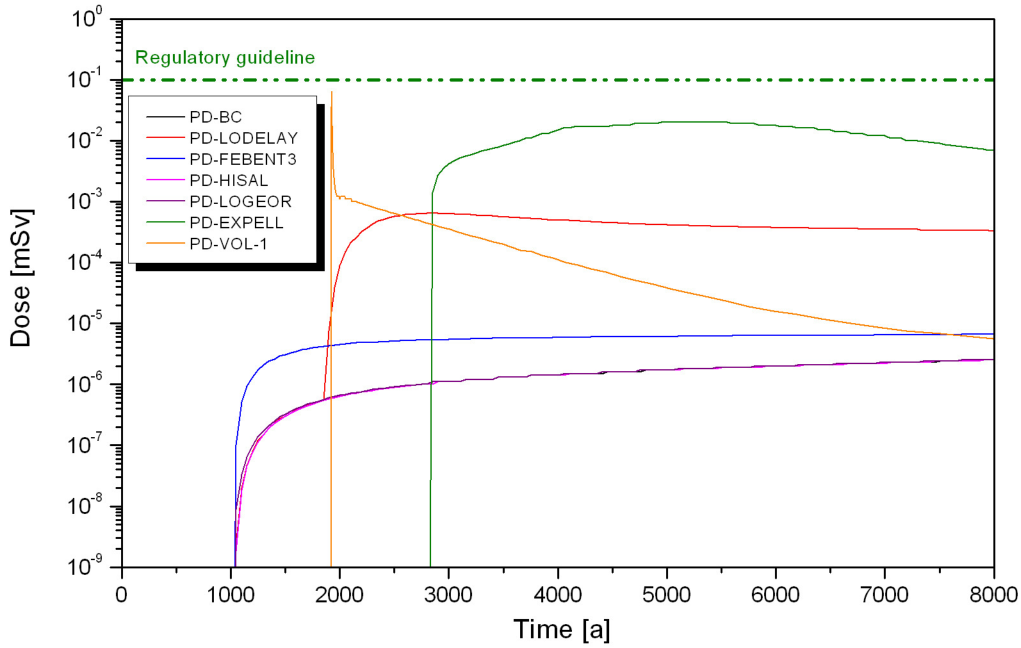


Figure 8-11. *Annual landscape dose to the most exposed individual due to potential releases from the repository in six most representative assessment cases for a canister with an initial penetrating defect (results for PD-BC, PD-HISAL and PD-LOGEOR approximately coincide).*

8.6.2 Single failed canister: the period after several thousand years

Figure 8-12 shows the overall geosphere release ratio maxima in all the calculation cases considered in the KBS-3H safety assessment (the individual cases are described in detail in the KBS-3H Radionuclide Transport Report, Smith et al. 2007c). The cases are arranged in two groups. At the bottom (the last four), are those cases where the geosphere release ratio maximum occurs in the first 10 000 years following repository closure, and for which the regulatory dose constraint is taken to apply. At the top are the geosphere release ratio maxima for the remaining cases. Within each group, cases are arranged in order of descending magnitude of the release ratio maximum.

The results show that the uncertainties with the largest impact on geosphere release ratio maxima, leading to maxima an order of magnitude or more in excess of the base case for a canister with an initial penetrating defect (PD-BC), are those associated with the possibility of:

- severe disruption of the buffer leading to canister failure by corrosion, especially in conjunction with a low assumed value for geosphere transport resistance (cases CC-GMW, CC-LOGEOR, CC-LOGEORG, CC-LOGEORS);
- disruption of the buffer-rock interface (cases PD-FEBENT1, PD-FEBENT2, PD-FEBENT3);

- expulsion of C-14 in volatile form by repository-generated gas through an initial penetrating defect (PD-VOL-1 and PD-VOL-2); and
- expulsion of contaminated water from the canister interior by repository-generated gas (PD-EXPELL).

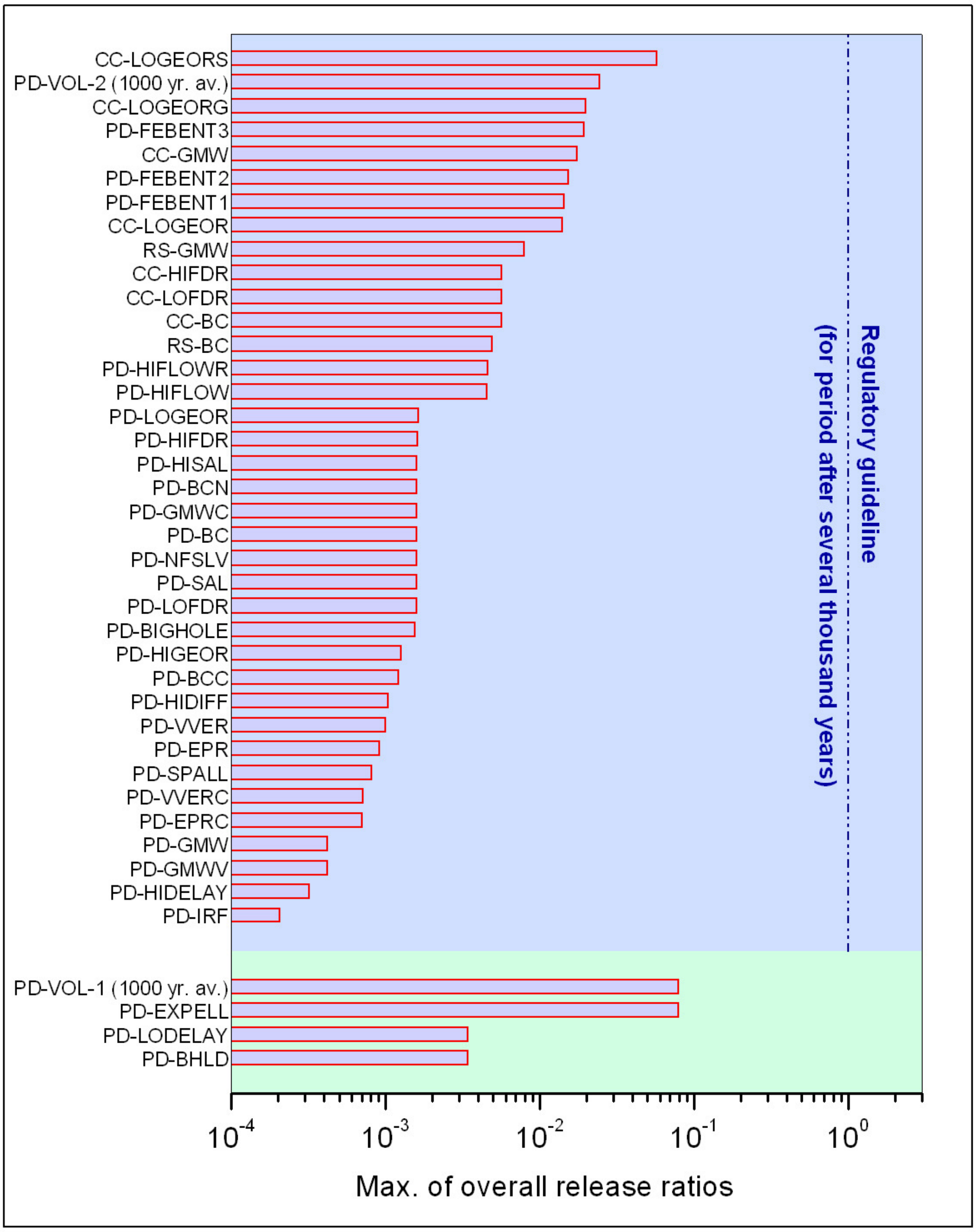


Figure 8-12. *Overall geosphere release ratio maxima in all calculation cases. Green background shading indicates the maxima that occur within the first 10 000 years post closure, which is interpreted in the present study as the "environmentally predictable future" (Smith et al. 2007a, Fig. 9-15).*

The highest calculated geosphere release ratio maximum occurs in case CC-LOGEORS, i.e. canister failure by copper corrosion, in association with an influx of glacial meltwater and loss of buffer mass by chemical erosion, coupled to an assumption of low transport resistance and saline geochemical conditions in the geosphere at later times. In this case, the geosphere release ratio maximum is more than an order of magnitude below the regulatory constraint. However, there are significant uncertainties associated with this scenario, e.g. whether substantial buffer mass loss by chemical erosion could occur at all, and, if it does, the number of canister positions that are likely to be affected (case CC-LOGEORS deals with only a single canister failure). The development of a better understanding of chemical erosion is a priority for future work for both the KBS-3H and KBS-3V repositories.

The next highest calculated geosphere release ratio maxima occur in cases PD-VOL-1 and PD-VOL-2, which address expulsion of C-14 in volatile form by repository-generated gas through an initial penetrating defect (see Section 5.9 of the Radionuclide Transport Report). Release ratios in these cases have been averaged over a 1 000 year interval, as allowed by Finnish regulations (note that such averaging is not allowed in the case of annual doses). Without such averaging, radionuclide release to the biosphere is close to the regulatory constraint, as discussed in Smith et al. (2007c).

It should be noted that in many cases assuming a canister with an initial penetrating defect, the geosphere release ratio maxima occur shortly after 10 000 years - i.e. beyond the period covered by the regulatory dose constraint - 10 000 years being the time assumed for loss of transport resistance of the defect in most cases. However, the uncertainty in the time when an initial penetrating defect loses its transport resistance is such that the maxima could equally well occur earlier - i.e. in the period up to several thousand years after repository closure - or much later, up to several hundreds of thousands of years in the future.

Partly to ensure that the 10 000 year assumption for loss of defect transport resistance did not result in potentially high doses being overlooked, WELL-2007 doses were calculated for all calculation cases across the entire million-year time frame covered by the KBS-3H safety assessment. The use of doses based on well scenarios is discussed in Section 4.4. The results, which are presented in Smith et al. (2007c), show that the calculated WELL-2007 dose maxima are also below the regulatory constraint of 0.1 mSv/a, although in most cases the maxima occur in the period after several millennia following closure, and so this constraint is not applicable to them. In those cases where the dose maxima occur in the environmentally predictable future when the regulatory dose constraint applies, the WELL-2007 dose maxima are 2-4 orders of magnitude below the constraint.

8.6.3 Likelihood or rate of canister failure

As in the case of the KBS-3V safety analysis described in Chapter 7, the KBS-3H safety analysis described in the previous sections considers the consequences of the failure of a single canister by different potential failure modes. Estimating the likelihood or rate of canister failure is an objective of ongoing work (see Ch. 11).

As described in Section 7.5.3, tentative estimates have been made of the number of canisters that might fail in the event of a large earthquake occurring in the vicinity of the repository. The failure of 16 of the 3 000 canisters in a KBS-3H repository is based on the conservative assumption that vulnerable locations are not avoided by applying rock suitability criteria. Based on the results given in Figure 8-12, even assuming that all 16 canisters in vulnerable locations fail, and do so simultaneously, the Finnish regulatory geo-bio flux constraint will still be met.

9 COMPLIANCE WITH REGULATORY REQUIREMENTS

This chapter (Fig. 9-1) discusses compliance with Finnish regulatory guidance on the long-term safety of geological disposal of spent fuel, as set out in the regulatory Guide YVL E.5.

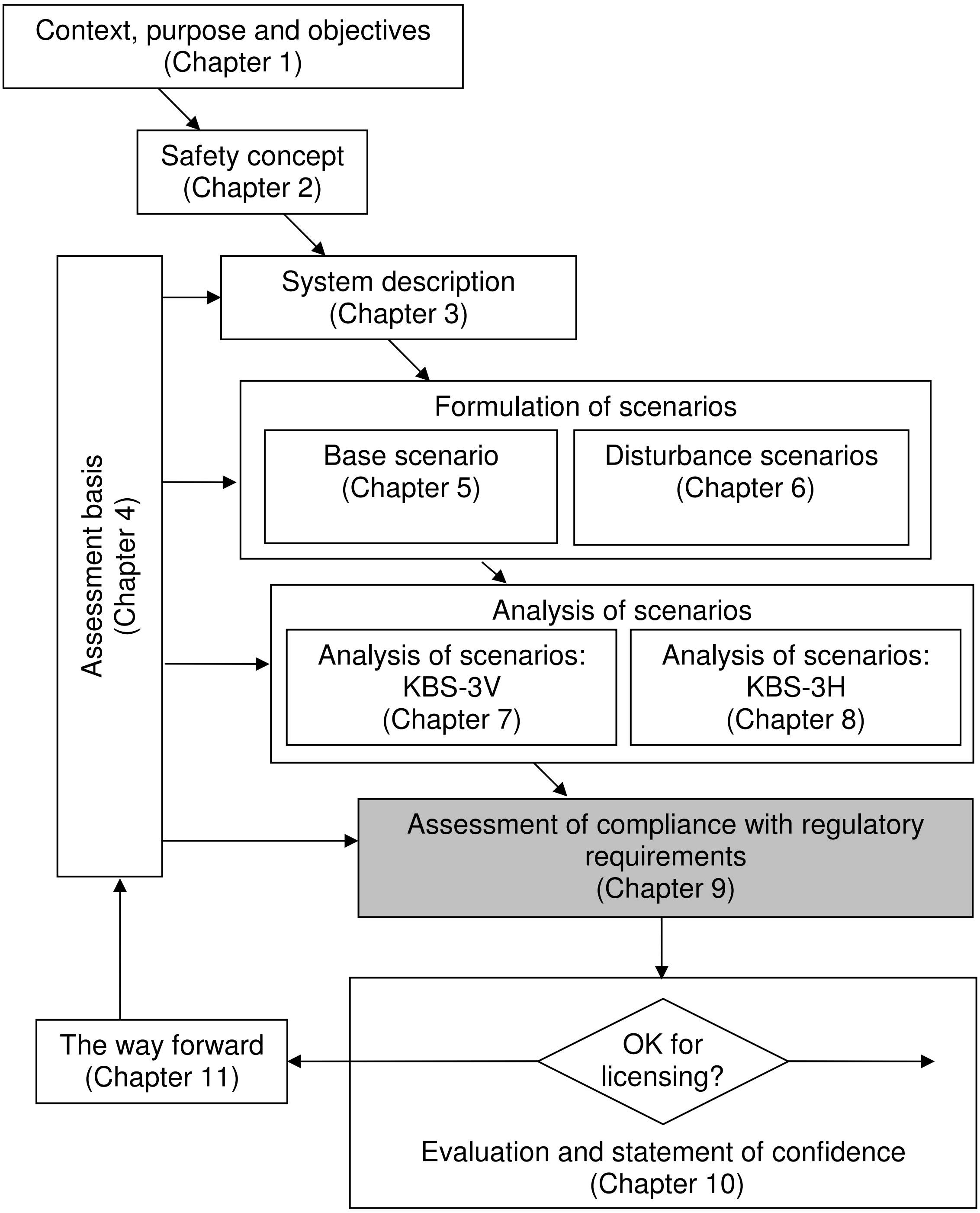


Figure 9-1. *The present chapter in the context of the safety case summary report.*

The chapter is structured as follows:

- Section 9.1 describes compliance with dose and release criteria; and
- Section 9.2 describes compliance with other requirements and guidance given in YVL E.5.

9.1 Dose and release criteria

Guide YVL E.5 specifies the annual dose and release criteria to be met by the repository, as summarised in Section 1.2.2. Compliance with these criteria has been described in Chapters 7 and 8 for both KBS-3V and KBS-3H repositories, where mainly single canister failure cases have been considered. To date, only in the case of canister failure due to rock shear have the numbers and consequences of multiple canister failures been estimated. The estimated geo-bio flux arising from multiple canister failures in this scenario, which conservatively disregards the application of rock suitability criteria to avoid fractures with the potential to undergo damaging shear movements (criteria are still under development), nevertheless complies with the regulatory geo-bio flux constraint.

Estimates of annual effective doses and of activity releases made in the KBS-3H and KBS-3V safety analyses are, with a high degree of certainty, considered to be overestimates, due to cautious assumptions incorporated throughout the assessment model chain that tend, for example, to underestimate barrier performance. This is consistent with YVL E.5 (Paragraph A1.8), which states:

"*...the selection of the computational methods and data shall be based on principle that the actual radiation exposures or quantities of released radioactive substances with high degree of certainty be lower than those obtained through safety analyses*."

Guide YVL E.5 (Paragraph 3.18) also gives the following guidance on the protection flora and fauna:

"Disposal shall not affect detrimentally to species of fauna and flora. This shall be demonstrated by assessing the typical radiation exposures of terrestrial and aquatic populations in the disposal site environment, assuming the present kind of living populations. The assessed exposures shall remain clearly below the levels which, on the basis of the best available scientific knowledge, would cause decline in biodiversity or other significant detriment to any living population."

Typical absorbed dose rates to identified assessment species have been estimated for releases to the surface environment calculated in the KBS-3V safety analysis. Following YVL E.5 (Paragraph 3.18), these dose rates are compared against internationally proposed screening values for the protection of biota against radiation in the environment. The particular values chosen are the organism group-specific screening values recommended by the PROTECT project (Andersson et al. 2008): 2 µGy/h for vertebrates 70 µGy/h for plants, and 200 µGy/h for invertebrates. The results show that the calculated typical absorbed dose rates are more than two orders of magnitude below the lowest proposed screening value. Thus, it is considered, with a high degree of confidence, that any releases from the repository do not affect species of flora and fauna detrimentally.

9.2 Other requirements and guidance

In addition to dose and release criteria, YVL E.5 also provides requirements and guidance related to repository design and performance targets, compliance with which is discussed in TKS-2009 (Posiva 2009a, Section 6.5.9). It also provides requirements and guidance on conducting and documenting safety analysis, compliance with which is described below.

According to the YVL E.5 (Paragraph 7.4), a safety analysis shall include:

1. A description of the disposal system and definition of the barriers and safety functions.

The disposal system comprises the repository, i.e. the system of engineered barriers and the surrounding bedrock, plus the overlying surface environment. A summary description of these elements and of the safety functions that are assigned to the main engineered components and to the bedrock is given in Chapter 3 of the present report. The description of the site is based on site descriptive model (SDM) reports (Posiva 2005, Andersson et al. 2007 and Posiva 2009b) and the Biosphere Description Report - BSD-2009 - (Haapanen et al. 2009a). The description of the engineered barriers is based on the canister design report by Raiko (2005) and on the repository layout and design reports for KBS-3V (Saanio et al. 2006, Kirkkomäki 2007) and for KBS-3H (Johansson et al. 2007; Autio et al. 2008).

2. A definition of the performance targets for the safety functions.

Preliminary performance targets have been defined for the engineered barriers, related to the capacity of these barriers to fulfil their safety functions. Preliminary bedrock target properties, related to the contribution of the bedrock to the performance of the engineered barriers and to retention of radionuclides within the bedrock, have also been defined as part of a set of rock suitability criteria (RSC). If the performance targets are achieved, and target properties are present, then the repository barriers are expected to fulfil their respective safety functions. According to YVL E.5 (Paragraph 4.7), performance targets are to be:

"based on high quality scientific knowledge and expert judgement".

The current performance targets and target properties, as given in TKS-2009 (Posiva 2009a, Section 6.1.4), are qualitative, but they have been given quantitative target values where appropriate, based on present-day scientific understanding. Performance targets and, in particular, the target values may be subject to change in the light of ongoing and future research, technical design and development work.

3. A definition of the evolutions describing the potential future behaviour of the disposal system.

According to YVL E.5 Paragraph 7.4,

"*Compliance with the long-term radiation protection requirements as well as the suitability of the disposal method and site shall be demonstrated by means of a safety case that shall include ..* "

amongst other components:

"*definition of the evolutions describing the potential future behaviour of the disposal system (scenario analysis)*"

In Posiva's terminology, the term "scenario formulation" is used in preference to "scenario analysis". This is to avoid confusion with the term "analysis of scenarios", which refers to the analysis of radionuclide release and transport in, and the radiological consequences of, scenarios involving radionuclide release from the repository. Evolution of the disposal system in the base scenario is described in Chapter 5 of the present report. Assessment scenarios are described in Chapter 6. These descriptions are based largely on the more detailed descriptions given in the Evolution Reports for a KBS-3V repository (Pastina & Hellä 2006) and for a KBS-3H repository (Smith et al. 2007b). According to Para. 3.16 of YVL E.5:

"Unlikely events induced by natural phenomena to be considered shall include a major rock movement in the vicinity of the repository".

The repository assessment scenarios described in Chapter 6 and analysed in Chapters 7 and 8 include the possibility of canister failure due to rock shear. Para. 3.16 of YVL E.5 also states that:

"Unlikely events caused by human actions to be considered shall include at least core-drilling hitting a disposal canister".

A quantitative analysis of the potential radiological consequences of human intrusion scenarios were outside the scope of both 2009 safety analysis of a KBS-3V repository and the KBS-3H safety assessment, although they have been considered in safety assessments of Swedish sites and in earlier safety analyses in Finland (Section 10.2.3). Human intrusion scenarios will also be formulated and analysed in future safety studies.

4. A functional description of the disposal system by means of conceptual and mathematical modelling and the determination of the input data needed in these models.

Modelling at different levels of detail is used to assess the impact of various processes and disruptive events on the capacity of the system to achieve its performance targets, and is a part of the methodology applied in the formulation of scenarios (Chapter 6).

5. An analysis of the quantities of radioactive substances that are released from the waste, penetrate the barriers and enter the biosphere, and analysis of the resulting radiation doses.

The analysis of activity releases and doses from radionuclides which are released from the waste, penetrate the barriers and enter to the surface environment is the main activity of safety assessment, and is described in Chapters 7 and 8 of the present report, based on the more detailed descriptions given for a KBS-3V repository in Nykyri et al. (2008) and in Hjerpe et al. 2010) and for a KBS-3H repository in the Radionuclide Transport Report (Smith et al. 2007c) and the Biosphere Analysis Report (Broed et al. 2007). According to Guide YVL E.5 (Paragraph 4.4), in accordance with the Government Decree on the safety of disposal of nuclear waste (DG 736/2008),

"*the long-term safety of disposal shall be based on safety functions achieved through mutually complementary barriers so that a deficiency of an individual safety function or a predictable geological change does not jeopardise the long-term safety.*"

The safety analyses have examined cases in which a canister fails by one of three possible modes: (i), an initial penetrating defect, (ii), corrosion (following perturbation of the buffer or buffer / rock interface or chemical erosion of the buffer by glacial meltwater) and (iii), rock shear. The results indicate that a deficiency in the performance of an individual canister does not jeopardise long-term safety, although the likelihood, timing and consequences of multiple canister failures remains an issue for further study (see Ch. 11).

Safety analyses have also examined a range of calculation cases in which the transport barrier performance of either the buffer or the host rock (or in some cases both) was less favourable than expected, and also in which release from the spent fuel matrix was more rapid than expected. Cases were, for example, examined in which the diffusion coefficient for the buffer was increased with respect to its value in the base calculation case, and also in which the rate of flow in the host rock was increased by one or more orders of magnitude. The uncertainties with the largest impact on geosphere release ratio maxima are those associated with the possibility of:

- severe disruption of the buffer leading to canister failure by corrosion, especially in conjunction with a low assumed value for geosphere transport resistance;
- disruption of the buffer-rock interface;
- expulsion of C-14 in volatile form by repository-generated gas through an initial penetrating defect; and
- expulsion of contaminated water from the canister interior by repository-generated gas.

The results indicate that the performance of none of the barriers is individually critical to long-term safety, provided the number of canister failures is limited.

6. Whenever practicable, estimation of the probabilities of activity releases and radiation doses arising from unlikely events impairing long-term safety.

Due to limitations in the understanding of relevant processes, only in the case of canister failure by rock shear following a large earthquake has an estimate so far been made of the likelihood or rate of canister failure. The better quantification of the likelihood or rate of canister failure is an objective of ongoing work (see Ch. 11).

7. Uncertainty and sensitivity analyses and complementary discussions.

Uncertainty and sensitivity analyses are central to recent safety analyses. Many such analyses were carried out in both the KBS-3V and KBS-3H safety analyses. The 2009 KBS-3V safety analysis identified and analysed a wide range of uncertainties. The KBS-3H safety assessment focussed mainly on identifying and analysing the impact of issues and uncertainties that were judged to have a different significance for, or potential impact on, KBS-3H compared with KBS-3V.

According to Guide YVL E.5, Paragraph A1.9:

"*The importance to safety of such scenarios that cannot reasonably be assessed by means of quantitative analyses shall be examined by means of complementary considerations. … Complementary considerations shall also be applied parallel to the actual safety assessment in order to enhance the confidence in results of the analysis or certain part of it*".

Complementary considerations are part of the evaluation of confidence described in Chapter 10 of the present report. These are based, in part, on material from the Complementary Evaluations of Safety Report (Neall et al. 2007). Some of the complementary considerations given in Neall et al. (2007), such as comparisons of elements of the safety case methodology with those used in other national programmes, as well as the main results and conclusions of other comparable safety assessments, will be presented in future in the *Analyses of Scenarios Report*.

8. Comparison of the outcome of analyses with the safety requirements.

Compliance with regulatory dose and release criteria in different time windows is described in Chapters 7 and 8.

9. Documentation of the safety case.

According to Finnish regulatory guidance (YVL E.5, Paragraph A1.10)):

"*The safety shall be documented carefully. The documentation shall aim at transparency, implying that each part of the safety, the basic assumptions, used methods, obtained results and coupling to wholeness case are evident. Another goal shall be traceability, implying that the justifications for the used assumptions, input data and models shall be easily found in the documentation.* "

The safety case portfolio was restructured in 2008, among other reasons, to satisfy these requirements for both transparency and traceability. In particular, the Summary Report is intended to present the evidence, arguments and analyses of the safety case in a manner that favours transparency. Other reports that describe the details of the safety assessment are intended to facilitate traceability (see Posiva 2008b).

10 EVALUATION AND STATEMENT OF CONFIDENCE

This chapter (Fig. 10-1) summarises the main evidence, arguments and analyses that lead to confidence in the good prospects that a geological repository for the final disposal of spent fuel, implemented as planned at the Olkiluoto site, will provide an adequate level of long-term safety.

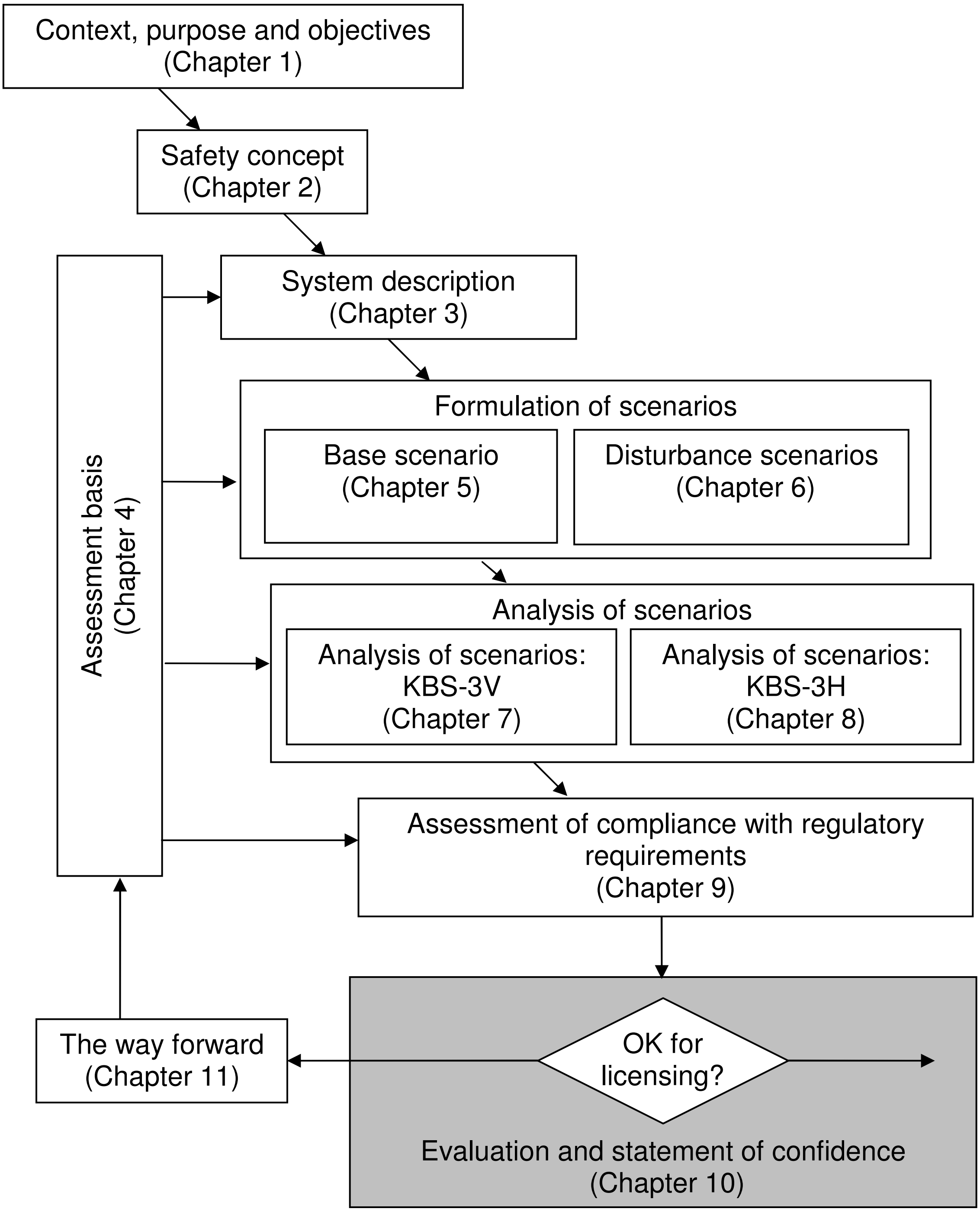


Figure 10-1. *The present chapter in the context of the safety case summary report.*

The chapter is structured as follows:

- Section 10.1 describes why the base scenario represents the expected evolution of most canisters over a time frame of several hundreds of thousands of years;
- Section 10.2 discusses confidence in the limited consequences of canister failure over this time frame;
- Section 10.3 discusses the consequences of the repository in the still longer term; and
- Section 10.4 provides a statement of confidence based on the available evidence, arguments and analyses.

10.1 Confidence in the base scenario

The present report outlines Posiva's preliminary safety case for the final disposal of Finnish spent fuel in a geological repository at the Olkiluoto site. An updated safety case will be developed to support the Preliminary Safety Assessment Report (PSAR) in 2012. However, studies to date already indicate that, except for a few unlikely circumstances affecting a small number of canisters, spent fuel is expected to remain isolated, and the radionuclides contained within the canisters, for hundreds of thousands of years or more, as described in the base scenario.

Confidence in the base scenario derives, in the first place, from the intrinsic properties of the main components of the repository that contribute to the safety functions and from the understanding of their evolution over time. This understanding has been gained from extensive site- and concept-specific field, laboratory and modelling studies and from studies, examples of which are given in Section 4.1.2, and from natural and anthropogenic analogues.

The canisters are mechanically strong and corrosion resistant (evidence from natural and anthropogenic analogues for the corrosion resistance of copper is mentioned below). They are also protected by the surrounding bentonite buffer and by their deep underground location in rock that is geologically very stable and lacks resources that might attract deep drilling activities in the future that could disturb the repository.

Conditions around the canisters and at the canister surface will change over the first few hundred years following repository closure, but these changes are not expected to have any significant effects on the integrity of the canisters themselves. For example, the bentonite clay buffer, which is initially only partly water saturated, will gradually take up water from the rock and will swell to fill any small gaps between the canisters and the rock that remain following emplacement of the buffer and canisters. This swelling will cause a pressure to be exerted on the canisters, but the canisters are designed to withstand such pressures with a large safety margin. Some oxygen from the atmosphere will be trapped within the repository following closure, but this will cause only very limited corrosion of the canister surfaces, and corrosion by oxygen will cease once all the oxygen in the repository is consumed by this and other chemical reactions.

The buffer will provide a protective environment for the canisters, particularly once saturated. In particular, the saturated buffer will be plastic and so will protect the canister from any small rock movements occurring following repository construction and, in the longer term, due to seismic activity at the site. Microbes that might otherwise produce chemical conditions that corrode the canisters, while present, will be virtually inactive in the saturated buffer. The saturated buffer will also provide a physical barrier that will greatly hinder chemical substances (particularly sulphide) that might corrode the canister from migrating from the rock to the canister surface.

In the longer term, conditions around the canisters will become much more stable than during early evolution. Corrosion of the canisters will continue to take place, due mainly to reaction with small amounts of sulphide that are present in the groundwater, but will be extremely slow. Canister failure by corrosion has been calculated to take millions of years, assuming that the buffer, as expected, provides a barrier that greatly hinders the migration of sulphide from the rock to the canister surface - even taking into consideration the possibility that, in certain locations, corrosion could occur more rapidly than on average.

Conditions underground will be affected to some extent by the formation of permafrost and ice sheets at the ground surface following a future change to much colder climatic conditions, although changes at repository depth will be far more limited than those closer to the surface. Glacial periods have occurred repeatedly in the past and are expected to occur in the future. However, the effects of ongoing human activities, particularly the release of greenhouse gases to the atmosphere, although uncertain, are likely to result in an unusually prolonged period of temperate, interglacial conditions. Numerical simulations indicate that, at Olkiluoto, the development of permafrost and frozen ground and the advance and retreat of ice sheets will have only limited effects on temperatures at repository depth, although the possibility of penetration of permafrost to repository depth cannot be completely ruled out. Major climate change could have a significant effect on the rate at which water seeps through the rock and on the chemical composition of the water, although these effects will be transient. Seismic activity, which is currently low at the Olkiluoto site, will be further suppressed by any future ice sheets. However, large earthquakes are possible when the ice sheets retreat. The layout and design of the repository, which includes a protective bentonite buffer around the canisters, are such that these changes and events are considered unlikely to affect most canisters to any significant extent.

Evidence and arguments to support this understanding of the long-term performance of the repository as a whole and of the materials present in the repository have been gained from extensive site- and concept-specific field and laboratory studies and from studies of natural and anthropogenic analogues. Ongoing studies are described in TKS-2009 (Posiva 2009a). Natural and anthropogenic analogues, which have included uranium ore deposits, native copper occurrences, copper and iron archaeological finds and deposits of bentonite and related clay materials, are discussed in Neall et al. (2007) and references therein.

10.2 Confidence in the limited consequences of canister failure

10.2.1 Barriers and features limiting radionuclide release and transport

Although the base scenario, in which canisters remain intact and no release of radionuclides occurs over hundreds of thousands of year or more, represents the expected evolution for most canisters in the repository over a time frame of at least several hundreds of thousands of years, a small number of canister failures within this period cannot be excluded.

The planned disposal system provides a series of barriers and processes that delay and attenuate the releases from a failed canister, such that any exposure of humans and other biota to radioactivity will occur only in the distant future and is highly unlikely to cause harm. The key barriers to radionuclide release and transport are:

- the copper-iron canister, which may, depending on the mode of failure, continue to provide some transport resistance for a period even after failure;
- the bentonite buffer;
- the backfill of the KBS-3V deposition tunnels; and
- the host rock.

The key features of these barriers contributing to the limited releases to the biosphere in the event of canister failure are:

- low flow rates of groundwater;
- low dissolution rates of spent fuel (under reducing chemical conditions) and low corrosion rates of fuel assembly materials;
- low solubilities of several of the most hazardous radionuclides;
- slow transport of radionuclides in the bentonite buffer (advection and colloidal transport avoided; sorption provided); and
- slow transport in the host rock (limited groundwater flow, diffusion into the rock matrix, sorption in the rock matrix).

10.2.2 Analysis of calculation cases

Confidence in safety in the event of canister failure is derived mainly from the results of the assessment of the radiological consequences of assessment scenarios in the 2009 safety analysis of a KBS-3V repository and in the KBS-3H safety assessment, and from the quality measures enacted to build confidence in these assessments.

A difference analysis carried out within the KBS-3H safety assessment has shown that the key differences in the evolution and performance of the KBS-3H and KBS-3V designs relate mainly to the engineered barrier system and to the impact of local variations in the rate of groundwater inflow on buffer saturation along the drifts. In the case of KBS-3H, particularly in tight drift sections, the gas generated by steel repository components external to the canister in the current design (principally the supercontainer shell) may accumulate at the buffer / rock interface, resulting in a prolonged period during which significant inflow of water from the surrounding rock will be limited and the buffer will remain only partially saturated. This is not considered to be a safety concern. However, the early evolution of the KBS-3H buffer, including the possibility of erosion by transient water flows (piping) during operations and subsequent saturation, drying and wetting, impact of iron saturation and cementation due to silica precipitation are all issues requiring more thorough investigation (some of these issues are also relevant to KBS-3V). The safety functions of the geosphere do not significantly differ between the two variants, although the importance of some geosphere properties may differ, e.g. the KBS-3H design is more sensitive to sub-vertical fractures with respect to potential damage to the engineered barrier system by rock shear.

Results of the safety analyses addressing the failure of a single canister indicate radiological consequences that are below regulatory constraints, generally by an order of magnitude or more, irrespective of identified uncertainties in canister failure mode and in release and transport processes. Although the rate or probability of canister failure has not as yet been quantified (except for preliminary estimates in the case of canister failure due to rock shear), this implies that, for any canister failure mode, ten or more canisters would have to fail at times that are sufficiently close together that the peak doses or releases are close to additive for the regulatory constraints to be exceeded. Differences in the geometry and transport paths considered in the analysis of the KBS-3V and KBS-3H design variants have only a minor impact on calculated releases and doses.

10.2.3 Analysis of human intrusion

Regarding human intrusion, although the geosphere lacks resources that might attract deep drilling activities in the future, scenarios involving drilling into the repository cannot be excluded. Calculation cases for post-closure human intrusion scenarios have not been developed or analysed in the Finnish programme since the TVO-85 safety analysis (Vieno et al. 1985), and so need to be addressed further in the future. Uncertainties in the nature of future human activities and the evolution of the state-of-the-art in science and technology are such that estimates of both probability and consequences of human intrusion scenarios must be based on "stylised assumptions" that cannot be fully substantiated or shown to be the most conservative. Regulations in Finland provide some guidance as to the types of intrusion scenarios to be considered. According to Guide YVL E.5 (Para. 3.16), unlikely events caused by human actions that are to be considered in safety assessment shall include:

"*… at least core-drilling hitting a disposal canister*".

Some indication of the consequences of damage to the Olkiluoto repository by future drilling can be obtained from safety assessments of Swedish sites, including, most recently, SR-Can. Here, it was assumed that drilling occurs 300 years after the sealing of the repository. Significant health consequences were shown to arise in the pessimistic case where material from the fuel elements is brought to the surface and left on the ground, and people spend time in the contaminated area. Note, however, that, according to Position Statement 9 of the German Working Group on Scenario Development: Internationale Zeitschrift für Kernenergie 2008:

"*With the decision for the concept of concentrating and isolating the radioactive waste in a repository, the possibility inevitably has to be accepted that radiation exposure limits may be exceeded in the event of intrusion into the repository … Therefore the Working Group holds the view that it is not reasonable to evaluate consequences of HI (human intrusion) by means of radiological limit values*."

Less significant was the case of the borehole being used as a well for drinking water and irrigation. The calculated annual effective doses in this latter case were calculated in SR-Can to be in the range of 0.1 to 1 mSv per year. This range is somewhat less, for example, than the typical natural external radiation exposure in Finland (around 3 mSv per year). Furthermore, in TVO-85, the probability of canister damage due to drilling was estimated to be very low: 2×10^{-8} per year. In Sweden, the estimated probability for the larger Swedish repository was 10^{-7} per year (SKB 1995). These low probabilities make the expectation value of dose very small for such scenarios.

The likelihood and consequence of post-closure human intrusion scenarios involving drilling are similar for KBS-3V and KBS-3H repositories, although there will be some small differences in the probability of a vertical borehole intersecting vertically compared with horizontally emplaced canisters. Computer modelling to calculate the consequences of human intrusion scenarios involving drilling will be carried out as part of the 2012 biosphere assessment.

10.2.4 Quality measures to enhance confidence in the analyses

Quality measures are being applied in the development and application of models, data and computer codes in the assessments, including:

- 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;
- validation of the models used to analyse the scenarios; and
- verification of assessment codes.

Table 10-1 gives some examples of current, planned and possible future validation of aspects of terrain and ecosystems development modelling (TESM) (Ikonen et al. 2010 presents a plan for testing of the TESM sub-models).

All computer codes used in Posiva's safety assessments are developed according to a quality assurance procedure and verified by comparison with analytical solutions, alternative codes and experimental data. Where possible, confidence in the modelling results is enhanced by means of the simulation of experiments and of natural analogue data. Posiva participates or has participated in international model validation studies such as the INTRAVAL (International Project to Study Validation of Geosphere Transport Models) project, which ended about 10 years ago, when the need for site-specific validation studies emerged.

Currently, benchmarking studies are used to compare different codes applied to the same system. For instance, the VTT-developed near-field release and transport code REPCOM has been verified against the code PORFLOW (Nordman & Vieno 2003) and the radionuclide release and transport module of the GoldSim simulation package (Pulkkanen 2009). In the KBS-3H safety assessment, a comparison was made between REPCOM results and results obtained using an alternative near-field code: the SPENT code used by Nagra in recent safety assessments in Switzerland (see Appendix A of Smith et al. 2007c). Posiva is also participating in international integration groups, such as NF-PRO (Near field Processes), which is exploring the use of different tools for modelling coupled thermo-mechanical-hydrological and chemical processes in deep geological disposal systems.

In the case of the biosphere codes, the predecessor of Pandora, Tensit, has been compared with several analytical results as well as numerical results from other simulation tools (Robinson et al. 2003), described in Jones et al. (2004). Pandora version 1 was successfully verified against one of the models, SN2, used in the Tensit test (Jones et al. 2004). Furthermore, Ecolego (Avila et al. 2003), which is a tool based on Matlab/Simulink using the same modelling approach as Pandora, has been compared to several other tools (Maul et al. 2003). The comparison of Ecolego with other tools contributes indirectly to confidence in Pandora.

Confidence in the analyses also derives from the systematic treatment of uncertainty. The current approach is described in Chapter 4, but is expected to be further developed in the future (Ch. 11).

Table 10-1. *Examples of current, planned and possible future validation of aspects of terrain and ecosystems development modelling (TESM).*

Land uplift model (past shorelines)	Comparison with available earlier estimates of Ancylus and Litorina stage shorelines do not reveal any significant differences
Peat growth	The results in TESM-2009 are reasonable in the light of known peat bog cross sections (Haapanen et al. 2009b)
Reed colonies	Model calibrated against survey results from Olkiluoto; comparison with independent data pending analysis of external datasets (Alahuhta 2008)
Forest type classification and parameters	Pending availability of updated national forest inventory data (results also dependent on the soil data available from the areas outside Olkiluoto where independent testing needs to be done)
Aquatic sedimentation and erosion	Needs detailed data to be able to evaluate the model results, collection phase ongoing

In future, the Models and Data Report of the safety case portfolio (Fig. 1-3) will have a central role in the quality management of the safety case, as will the introduction of a new expert elicitation procedure. The Models and Data Report will systematically document the models, data and computer codes used in safety analyses. The version foreseen for 2012 will include the whole production chain for models and data used in the safety case, including quality measures such as those listed above.

10.2.5 Analysis results in perspective

The calculated low radionuclide release rates to the biosphere and resultant dose maxima imply that any radiological consequences of these releases will be negligible. To place the calculated doses in perspective, the typical natural external radiation exposure in Finland is around 0.4 to 3 mSv per year (0.05 to 0.3 μSv per hour; STUK 2007), and the average Finnish exposure to all ionising radiation is around 4 mSv per year. This includes both natural and man-made sources of radiation, such as medical x-rays and the Chernobyl fall-out. Finnish regulations specify a constraint of 0.1 mSv per year for the dose received by the most exposed members of the public over the first several thousand years after repository closure. Calculated dose maxima are well below these levels; the annual landscape dose maxima to a representative person for the most exposed group calculated in the 2009 safety analysis of a KBS-3V repository was below 10^{-4} mSv in all cases. Further comparisons of calculated doses with doses due to other sources of exposure are given in Neall et al. (2007).

In addition to calculated radiation doses, other safety indicators are also evaluated Neall et al. (2007). For example, the radiotoxicity flux is a measure of the hazard created by a flux of radionuclides across a defined interface within a given period. For these evaluations, a hypothetical interface of 1 km^2 above the repository was considered. The total calculated annual flux from a KBS-3H repository through the geosphere to the biosphere was assumed to pass across this interface. This simple construction allows the repository releases to be compared with a range of naturally occurring radionuclide fluxes (Fig. 10-2); see Appendix B of Neall et al. (2007) for further explanation.

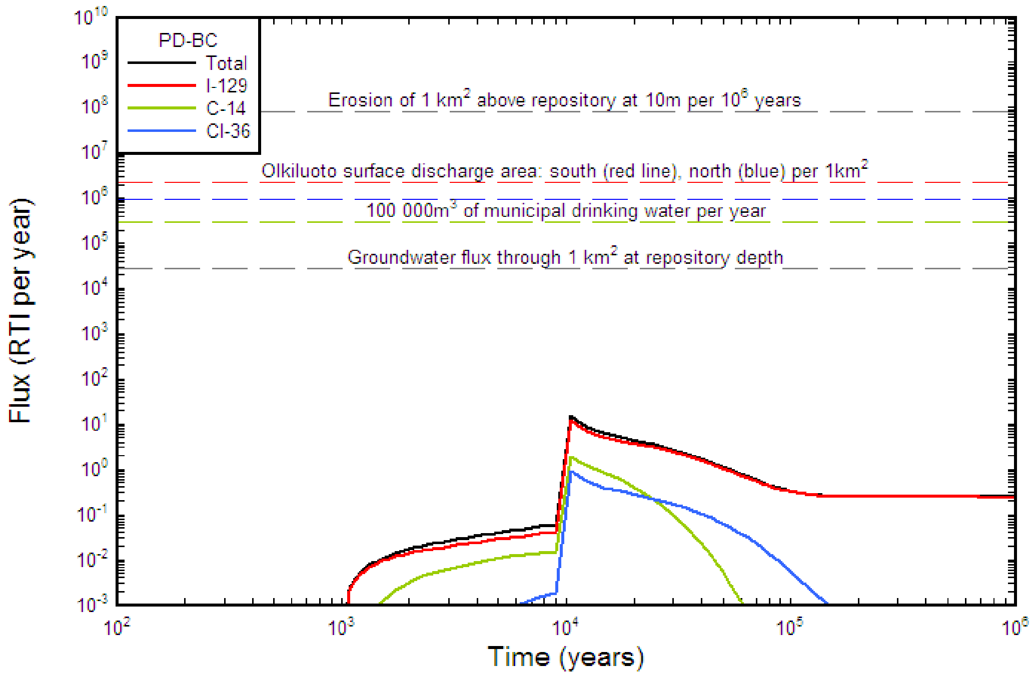


Figure 10-2. *Radiotoxicity flux from the KBS-3H repository into the biosphere for the penetrating defect base calculation case (case PD-BC in the KBS-3H safety assessment) compared with a range of naturally occurring radiotoxicity fluxes (Neall et al. 2007, Fig. 6-4).*

These evaluations confirm the insignificance of the calculated repository releases when compared with natural radiotoxicity fluxes associated with groundwater discharge in the Olkiluoto area, or erosion of the not-particularly uranium-rich rocks in the area.

10.3 Consequences in the farthest future

Although the possibility of some earlier canister failures cannot be completely excluded, it is expected that at least most of the canisters in the repository will remain intact and provide complete containment of all radionuclides associated with the fuel for a period of a several hundreds of thousands of years or more.

Beyond this time frame, slow corrosion of the copper shell, the detrimental effects of multiple periods of glaciation, or some other mechanism will eventually lead to failure of all the canisters and the release of some radioactivity to the surrounding rock. The radioactivity that is initially present in the repository will decay to a much reduced level before this happens. Some limited releases to the surface environment are likely to occur, but there are large uncertainties in quantitative safety analyses over such long time periods, so that other forms of analysis and argumentation are judged to be more appropriate. This is recognised in Finnish regulations. According to Guide YVL E.5 (Para. A1.9):

"… *safety evaluations extending beyond a time horizon of once million years can be based mainly on the complementary considerations.*"

where these may include:

"*… e.g. analyses by simplified methods, comparisons with natural analogues or observations of the geological history of the disposal site*".

Such analyses and lines of argument will be developed in the Complementary Considerations Report of the safety case portfolio.

At Olkiluoto, there is evidence of regional stability over millions of years, with no indication that this situation will be disrupted by changes in plate tectonics in the next few million years. However, after hundreds to thousands of millions of years, slow geological processes could result in the rock being gradually lifted and eroded, so that repository materials eventually reach the surface.

In the hundreds to thousands of millions of years before the repository horizon is exposed to the surface by successive cycles of erosion and uplift, the evolution of the repository materials is uncertain and any comments are necessarily speculative. In some respects, after very long times the repository materials may tend to resemble a heterogeneous uranium ore body, perhaps analogous to granite- or sediment-hosted Cu-U deposits. The isotopic composition of a natural ore body will, however, differ from that of spent fuel. In particular, some of the artificial radionuclides produced in the reactor will still be present in the far future, and their implications for safety will need to be assessed.

10.4 Statement of confidence

A full safety case for the KBS-3V variant of the KBS-3 method will be carried out to support the Preliminary Safety Assessment Report (PSAR) in 2012. The KBS-3H alternative will be analysed using a common methodology and the cases studied will differ only to the extent that this is required by the different designs. A full safety case for the KBS-3H alternative will be carried out after PSAR. Already, however, it can be concluded based on the safety assessment carried out to date, that both variants for a repository located at Olkiluoto show good promise from the long-term safety point of view.

In both variants, safety is provided, in the first place, by mechanically strong and corrosion resistant canisters, emplaced deep underground in a stable geological formation and surrounded by a protective bentonite clay buffer. Except for a few unlikely circumstances affecting a small number of canisters, spent fuel is expected to remain isolated, and the radionuclides contained within the canisters, for hundreds of thousands of years or more.

The results of the assessment of scenarios confirm that releases from a failed canister will be low and cause no significant harm to humans and other biota. Only single canister failure cases have, however, so far been considered and the possibility of multiple canister failures must be addressed in future studies. Nevertheless, in most of the calculation cases considered, the substantial margins by which the regulatory constraints for radionuclide release and dose are met indicate that even the occurrence of multiple failed canisters will not undermine the safety of either a KBS-3V or KBS-3H repository.

The main residual issues and uncertainties are identified in the most recent safety assessments. A strategy to adequately manage these is an essential part of the safety case. Current plans are set out in TKS-2009 (Posiva 2009a), and some key aspects are discussed further in Chapter 11.

11 THE WAY FORWARD

This chapter (Fig. 11-1) describes the broad strategy for identifying and managing safety-related issues and uncertainties that will have to be addressed prior to the compilation of the Preliminary Safety Analysis Report (PSAR) and the Final Safety Analysis Report (FSAR), and lists the main issues.

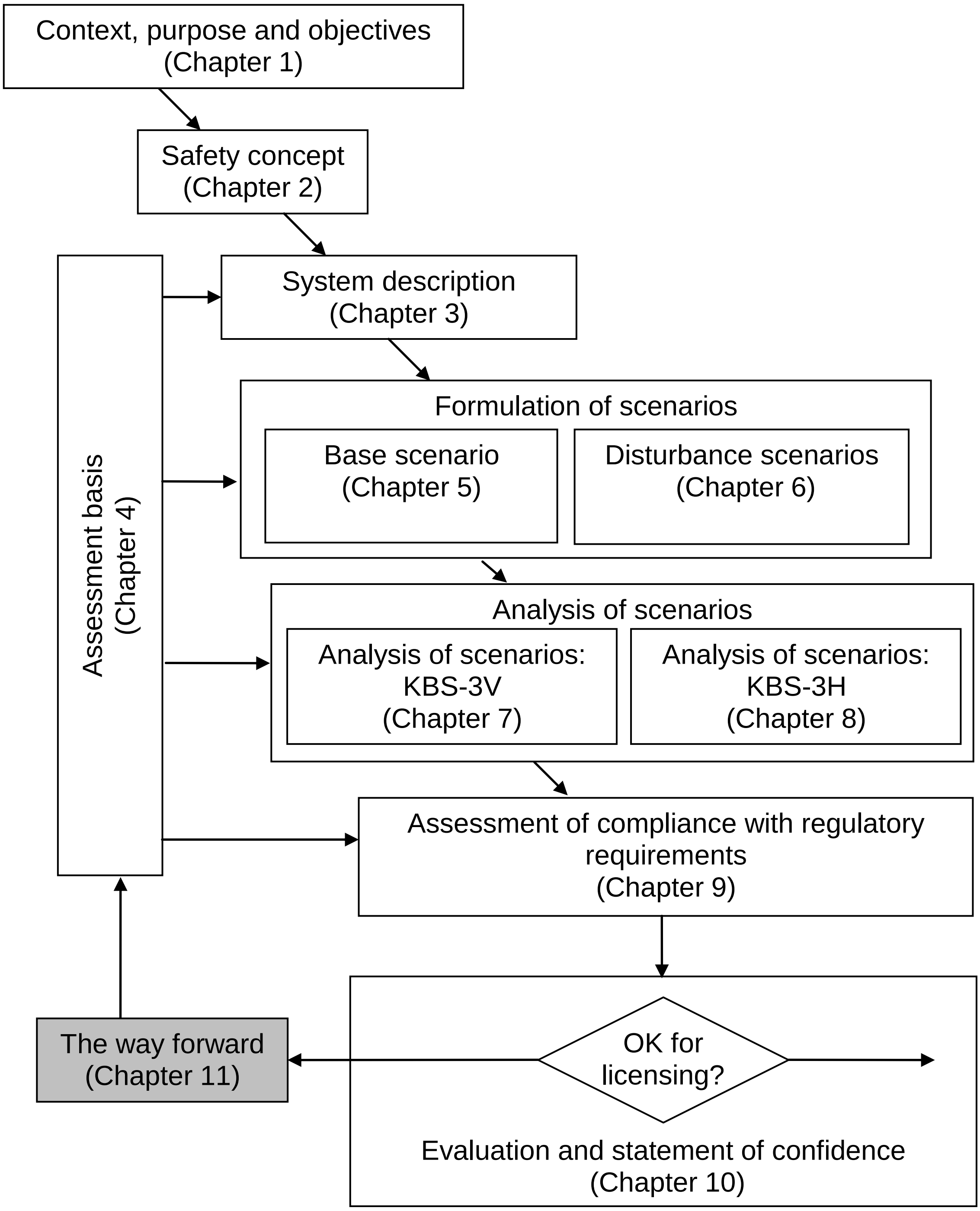


Fig. 11-1. *The present chapter in the context of the safety case summary report.*

As noted in Section 1.1.4, the development of the disposal system is based on the idea of robustness, which means, where practicable, avoiding concepts and components the behaviour of which is difficult to understand and predict, and reducing the impact of uncertainties – for example by introducing conservative safety margins in the design of some components.

The establishment of performance targets for the main system components provides guidance to robust system design, e.g. the range of saturated buffer densities that ensure that the performance targets for the buffer are achieved, and hence that the buffer safety functions are fulfilled. Safety assessment considers the impact of uncertainties in the assumptions underlying the performance targets and on the capacity of the system to meet the targets as it evolves over time. In the safety analyses described in the present report, the deterministic approach to the identification and evaluation of calculation cases allows the impact of uncertainties on system performance and safety to be determined individually. This allows uncertainties that could potentially weaken the safety case to be identified, and avoided or reduced by research, technical design and development. A more systematic analysis of combinations of parameter uncertainties will be included in future safety assessments through a combination of probabilistic and deterministic approaches.

Current plans to address specific safety-related issues are given in TKS-2009 (Posiva 2009a), including issues regarding safety case development as well as more specific issues of scientific understanding. Key issues for safety case conceptualisation and methodological development include:

- how to handle the possibility of multiple canister failures;
- how to handle the combined effects of more than one disruptive event or process;
- how to ensure all relevant uncertainties are identified and their impacts assessed, including the potential use of probabilistic methods; and
- how to assure quality in the selection of models and data and, more generally, in the various steps in the production of the safety case.

If multiple canister failures occur at around the same time, or if geosphere release rates maxima are spread over a prolonged time interval, then, to a first approximation, the overall release maximum is obtained by multiplying the release from one failed canister by the total number of failed canisters. However, conditions affecting radionuclide transport may vary widely between canister positions and so, in principle, spatial heterogeneity needs to be taken into account when combining the releases from multiple failed canisters. Information on the spatial variability of groundwater flow and geosphere transport resistance can be extracted from the results of groundwater flow modelling.

A key issue to be addressed in future safety assessments is the systematic tracing of uncertainties from their source to their handling in calculation cases. The Models and Data Report will play a key role in this regard. In future safety assessments, the emphasis is likely to remain on deterministic calculations to illustrate the effects of individual uncertainties or uncertainties in combination. Thus, uncertainties that could potentially weaken the safety case can be identified, and avoided or reduced by research, technical design and development. However, a "probabilistic" approach, in which parameter values are sampled randomly from probability distributions (or probability density functions, PDFs), may also be considered in conjunction with deterministic calculations. A deterministic approach, however, can give a clear illustration of the impact of specific uncertainties. Furthermore, it avoids the need to define PDFs that quantify in single distributions widely different types of uncertainty (e.g. "aleatory" uncertainties related to variability or randomness and "epistemic" uncertainties arising, for example, where there is a range of plausible alternative models consistent with current scientific knowledge). However, probabilistic calculations provide a systematic approach to exploring all possible parameter combinations, and also lend themselves to the analyses of scenarios where well substantiated PDFs are available, and the calculated release rates to the biosphere or dose can be expressed in terms of expectation values for comparison with regulatory constraints, as allowed by Finnish regulations (YVL E.5, Paragraph 3.17). Nevertheless, the main emphasis will remain on deterministic analyses, complemented by scoping calculations to address, in particular, the extent to which potentially detrimental processes will affect the capacity of the system components to meet their performance targets or target properties.

Models and data used in the 2009 safety analysis of a KBS-3V repository and in the KBS-3H safety assessment were selected based on the preliminary information available at the time of carrying out these assessments. A more formal procedure is foreseen for future work (Posiva 2008b), which will be documented, along with the models and data selected and an analysis of model and data uncertainties, in the Models and Data Report (Fig. 1-3) in 2012. In particular, a more formal quality review of models and data will be undertaken. All steps in the production of models and data will be considered, e.g. starting with observations and measurements, continuing with primary interpretations and necessary abstractions (inference, upscaling, expert judgement) and ending with the derivation of effective parameters or datasets that can be used in safety assessment models and calculations. Each of these steps will have its own quality control and assurance measures. A specific area of methodological development with quality assurance aspects is expert elicitation. Plans for use of a more formal elicitation process are described in the Safety Case Plan 2008 (Posiva 2008b, Section 5.6.1).

Currently, comparisons between the analyses of assessment scenarios for KBS-3V and KBS-3H are complicated by the differences in the assumptions in the calculation cases considered and in the models and data used. For the PSAR, the scenarios for both variants will be analysed using a common methodology and the cases studied will differ only to the extent that this is required by the different designs.

Other areas of conceptualisation and methodological development include the analysis of human intrusion scenarios, which are required to be considered according to Finnish regulations, and the analysis of the safety implications of significant deviations or major mishaps or accidents during construction or operation of the repository. Additional biosphere assessment scenarios will also be directly developed based on the regulatory guidelines (YVL E.5). Development is foreseen in the modelling of the geosphere/biosphere interface. In particular, in forthcoming assessments the surface and near-surface hydrological model will be used to derive the future groundwater pressure head boundary condition for the EPM and DFN groundwater flow simulations, thus ensuring a fully consistent definition of the groundwater system in respect of both the infiltration and discharge of the deep groundwater and the flow paths of releases from failed canisters.

Key issues of scientific understanding relate to the evaluation of the likelihood or rate of canister failure by different modes. Scenarios giving rise to canister failure include the possibility of initial defects, and, for a few canisters in less favourable locations, damage by large earthquakes, which are most likely subsequent to any future period of glaciation, and failure by corrosion following misemplacement or later loss of the protective bentonite clay buffer, for example by chemical erosion following exposure to glacial meltwater. Better understanding of these scenarios is a key area of ongoing research, technical design and development. Note that, according to YVL E.5 (Paragraph 3.17), demonstration of compliance can take account of the low probability of some scenarios, provided their probability of occurrence can be estimated:

"The importance to safety of such incidental event shall be assessed and whenever practicable, the resulting annual dose or activity release shall be calculated and multiplied by the estimated probability of its occurrence."

The base scenario currently includes all lines of evolution of the disposal system giving no release of radionuclides. In future, those lines of evolution that lead to radionuclide release and that cannot be shown to have a low probability of occurrence may be included either in the base scenario or in variant scenarios, in accordance with the terminology for scenario classification given in YVL E.5.

As indicated above, the processes that are potentially the most detrimental to repository safety are related to glacial conditions (this was also a conclusion of the Swedish SR-Can safety assessment). The timing of any future period of glaciation is uncertain as is the climatic evolution in general. As described in TKS-2009 (Posiva 2009a, Section 6.5.1), climate model development will be undertaken to better estimate the time windows for warm and cold periods, their probabilities and possible extremes, in different climate scenarios over the next 100 000 years.

A number of other issues remain to be addressed before construction, operation and closure of the final repository. Many are relevant to both KBS-3V and KBS-3H. These include, for example, site-specific issues such as the transport rate of abiogenic methane and the kinetics of sulphate reduction in the rock. While some issues, such as those related to gas generation prior to canister failure, are relevant mainly to KBS-3H, it should also be noted that there are some issues that are specific to KBS-3V, such as those related to the deposition tunnel, its backfill and its excavation damaged zone.

Uncertainties cannot be completely eliminated by research, technical design and development. Rather, the aim is to design a repository that is sufficiently robust so that these residual uncertainties do not to affect long-term safety. In this context, demonstrating the feasibility and quality of technical solutions by tests and experiments is essential, and are planned as part of the quality assurance of the various production steps of the safety case. In this way, a comprehensive safety case will be developed to support the licensing process.

REFERENCES

Ahokas, H., Hellä, P., Ahokas, T., Hansen, J., Koskinen, K., Lehtinen, A., Koskinen, L., Löfman, J., Mészaros, F., Partamies, S., Pitkänen, P., Sievänen, U., Marcos, N., Snellman, M. & Vieno, T. 2006. Control of water inflow and use of cement in ONKALO after penetration of fracture zone R19. Posiva Working Report 2006-45. Posiva Oy, Olkiluoto, Finland.

Ahonen, L. & Vieno, T. 1994. Effects of glacial meltwater on corrosion of copper canisters. YJT Report YJT-95-19. Nuclear Waste Commission of Finnish Power Companies (YJT), Helsinki, Finland.

Ahjos, T. & Uski, M. 1992. Earthquakes in northern Europe in 1375-1989. Tectonophysics, 207, 1–23.

Alahuhta, J. 2008. Selkämeren rannikkovesien tila, vesikasvillisuus ja kuormitus (in Finnish; State, aquatic flora and loading of the coastal waters of the Bothnian Sea). Lounais-Suomen ympäristökeskuksen raportteja, vol. 2008, 9, 111 p. ISBN 978-952-11-3062-5. http://www.ymparisto.fi/

Andersson, J., Ahokas, H., Hudson, J. A., Koskinen, L., Luukkonen, A., Löfman, J., Keto, V., Pitkänen, P., Mattila, J. & Ikonen, A. T. K. 2007. Olkiluoto site description 2006. Posiva Report POSIVA 2007-03. Posiva Oy, Eurajoki, Finland.

Andersson, P., Beaugelin-Seiller, K., Beresford, N. A., Copplestone, D., Della Vedova, C., Garnier-Laplace, J., Howard, B. J., Howe, P., Oughton, D.H., Wells, C. & Whitehouse, P. 2008. Numerical benchmarks for protecting biota from radiation in the environment: proposed levels, underlying reasoning and recommendations. PROTECT (Protection of the Environment from Ionising Radiation in a Regulatory Context), EC Project, FI6R-036425.

Andersson, M., Ervanne, H., Glaus, M. A., Holgersson, S., Karttunen, P., Laine, H., Lothenbach, B., Puigdomenech, I., Schwyn, B., Snellman, M., Ueda, H., Vuorio, M., Wieland, E. & Yamamoto, T. 2009. Development of methodology for evaluation of long-term safety aspects of organic cement paste components. Posiva Working Report 2008-28. Posiva Oy, Eurajoki, Finland.

Anttila M. 2005. Radioactive characteristics of the spent nuclear fuel of the Finnish nuclear power plants. Posiva Working Report 2005-71. Posiva Oy, Eurajoki, Finland.

Anttila, P., Ahokas, H., Front, K., Heikkinen, E., Hinkkanen, H., Johansson, E., Paulamäki, S., Riekkola, R., Saari, J., Saksa, P., Snellman, M., Wikström, L. & Öhberg, A. 1999. Final disposal of spent nuclear fuel in Finnish bedrock - Olkiluoto site report. Posiva Report POSIVA 99-10. Posiva Oy, Helsinki, Finland.

Åstrand, P.-G., Jones, J., Broed, R. & Avila, R. 2005. PANDORA technical description and user guide. Posiva Working Report 2005-64. Posiva Oy, Eurajoki, Finland.

Autio, J., Börgesson, L., Sandén, T., Rönnqvist, P.-E., Berghäll, J., Kotola, R., Parkkinen, I., Johansson, E., Hagros, A. & Eriksson, M. 2007. Design description 2006. Posiva Working Report 2007-105. Posiva Oy, Eurajoki, Finland or SKB Report R-08-32. Swedish Nuclear Fuel and Waste Management Co (SKB), Stockholm, Sweden.

Autio, J., Anttila, P., Börgesson, L., Sandén, T., Rönnqvist, P.-E., Johansson, E., Hagros, A., Eriksson, M., Halvarsson, B., Berghäll, J., Kotola, R. & Parkkinen, I. 2008. KBS-3H Design description 2007. Posiva Report POSIVA 2008-01. Posiva Oy, Eurajoki, Finland or SKB Report R-08-44. SKB. Swedish Nuclear Fuel and Waste Management Co (SKB), Stockholm, Sweden.

Avila, R., Broed, R. & Pereira, A. 2003. Ecolego - A toolbox for radioecological risk assessment. International Conference on the Protection of the Environment from the Effects of Ionising Radiation, 6–10 October 2003, Stockholm, Sweden.

Börgesson, L. & Hernelind, J. 1999. Coupled thermo-hydro-mechanical calculations of the water saturation phase of a KBS-3 deposition hole - Influence of hydraulic rock properties on the water saturation phase. SKB Technical Report TR-99-41. Swedish Nuclear Fuel and Waste Management Co (SKB), Stockholm, Sweden.

Börgesson, L. & Hernelind, J. 2006a: Canister displacement in KBS-3V repository - a theoretical study. SKB TR 06-04, Swedish Nuclear Fuel and Waste Management Co (SKB), Stockholm, Sweden.

Börgesson, L. & Hernelind, J. 2006b. Consequences of loss or missing bentonite in a deposition hole. SKB Technical Report TR-06-13. Swedish Nuclear Fuel and Waste Management Co (SKB), Stockholm, Sweden.

Börgesson, L. & Sandén, T. 2006. Piping and erosion in buffer and backfill materials. Current knowledge. Studies of buffers behaviours in KBS-3H concept. SKB Report R-66-80. Swedish Nuclear Fuel and Waste Management Co. (SKB), Stockholm, Sweden.

Börgesson, L., Sandén, T., Fälth, B., Åkesson, M. & Lindgren, E. 2005. Studies of buffers behaviours in KBS-3H concept. SKB Report R-05-50. Swedish Nuclear Fuel and Waste Management Co. (SKB), Stockholm, Sweden.

Börgesson, L., Fälth, B. & Hernelind, J. 2006. Water saturation phase of the buffer and backfill in the KBS-3V concept. Special emphasis given to the influence of the backfill on the wetting of the buffer. SKB Technical Report TR-06-14. Swedish Nuclear Fuel and Waste Management Co (SKB), Stockholm, Sweden.

Bradbury, M. & Baeyens, B. 2003. Near-field sorption databases for compacted MX-80 bentonite for performance assessment of a high-level radioactive waste repository in Opalinus Clay host rock. Technical Report 02-18. Nagra, Wettingen, Switzerland.

Broed, R. 2007. Landscape model configuration for biosphere analysis of selected cases in TILA-99 and in KBS-3H safety evaluation, 2007. Posiva Working Report 2007-108. Posiva Oy, Eurajoki, Finland.

Broed, R., Avila, R., Bergström, U., Hjerpe, T. & Ikonen, A. 2007. Biosphere analysis for selected cases in TILA-99 and in KBS-3H safety evaluation, 2007. Posiva Working Report 2007-109. Posiva Oy, Eurajoki, Finland.

Buoro, A., Dahlbo, K., Wiren, L., Holmen J., Hermanson, J., Fox, A. 2009. Geological Discrete-Fracture Network Model (version 1) for the Olkiluoto Site, Finland. Eurajoki, Finland: Posiva Oy. 158 p. Working Report 2009-77.

Cedercreutz, J. 2004. Future climate scenarios for Olkiluoto with emphasis on permafrost. Posiva report POSIVA 2004-06. Posiva Oy, Eurajoki, Finland.

Dillström, P. 2005. Probabilistic analysis of canister inserts for spent nuclear fuel. Swedish Nuclear Fuel and Waste Management Co (SKB), Stockholm, Sweden. SKB Technical Report TR-05-19.

Enescu, N., Cosma, C. & Balu, L. 2003. Seismic VSP and crosshole investigations in Olkiluoto, 2002. Posiva Working Report 2003-13. Posiva Oy, Eurajoki, Finland.

FTRANS, 1983. A two-dimensional code for simulating fluid flow and transport of radioactive nuclides in fractured rock for repository performance assessment. Report ONWI-426. Intera Environmental Consultants Inc., Houston, USA.

Grambow, B., Lemmens, K., Minet, Y., Poinssot, C., Spahiu, K., Bosbach, D., Cachoir, C., Casas, I., Clarens, F., Christiansen, B., de Pablo, J., Ferry, C., Giménez, J., Gin, S., Glatz, J.P. , Gago, J.A., Gonzalez Robles, E., Hyatt, N.C., Iglesias, E., Kienzler, B., Luckscheiter, B., Martinez-Esparza, A., Metz, V., Ödegaard-Jensen, A., Ollila, K., Quiñones, J., Rey, A., Ribet, S., Rondinella, V.V., Skarnemark, G., Wegen, D., Serrano-Purroy, D. & Wiss, T. 2008. Final Synthesis Report, NF-PRO, RTD Component 1: Dissolution and release from the waste matrix. Deliverable D1.6.3. European Commission 2008.

Gribi, P., Johnson, P., Suter, D., Smith, P., Pastina, B. & Snellman, M. 2007. Safety assessment for a KBS-3H spent nuclear fuel repository at Olkiluoto - Process Report. Posiva Report POSIVA 2007-09. Posiva Oy, Eurajoki, Finland or SKB Report R-08-36. Swedish Nuclear Fuel and Waste Management Co. (SKB), Stockholm, Sweden.

Grimwood, P. & Thegerström, C. 1990. Assessment of the risks associated with human intrusion at radioactive waste disposal sites – Some observations from an NEA workshop. Proceedings of the symposium on safety assessment of radioactive waste repositories, Paris, NEA/IAEA/CEC, 385-395.

Grivé, M., Montoya, V. & Duro, L. 2007. Determination and assessment of the concentration limits for radionuclides for Posiva. Posiva Working Report 2007-103. Posiva Oy, Eurajoki, Finland.

Gunnarsson, D., Keto, P., Morén, L. & Sellin, P. 2007. Deep repository – backfill and closure. Assessment of backfill materials and methods for deposition tunnels. Posiva Working Report 2006-64. Posiva Oy, Eurajoki, Finland.

Haapanen, R., Aro, L., Ilvesniemi, H., Kareinen, T., Kirkkala, T., Lahdenperä, A.-M., Mykrä, S., Turkki, H. & Ikonen, A.T.K. 2007. Olkiluoto biosphere description 2006. Posiva Report POSIVA 2007-02. Posiva Oy, Eurajoki, Finland.

Haapanen, R., Aro, L., Helin, J., Hjerpe, T., Ikonen, A.T.K., Kirkkala, T., Koivunen, S., Lahdenperä, A.-M., Puhakka, L., Rinne, M. & Salo, T. 2009a. Olkiluoto Biosphere Description 2009. POSIVA 2009-02; www.posiva.fi.

Haapanen, R., Aro, L., Ikonen, A.T.K., Kirkkala, T., Koivunen, S., Lahdenperä, A-M. & Paloheimo, A. 2009b. Potential reference mires and limnic ecosystems for biosphere assessment of Olkiluoto site. Posiva Working Report (In preparation). Posiva Oy, Eurajoki, Finland.

Hagros, A. 2007a. Estimated quantities of residual materials in a KBS-3H repository at Olkiluoto. Posiva Working Report 2007-104 Posiva Oy, Eurajoki, Finland or SKB Report R-08-33, Swedish Nuclear Fuel and Waste Management Co (SKB), Stockholm, Sweden.

Hagros, A. 2007b. Foreign materials in the repository – Update of estimated quantities. Eurajoki, Finland. Posiva Working Report 2007-17. Posiva Oy, Eurajoki, Finland.

Hakala, M., Hudson, J. A., Harrison, J. P. & Johansson, E. 2008. Assessment of the potential for rock spalling at the Olkiluoto Site. Posiva Working Report 2008-83. Posiva Oy, Eurajoki, Finland.

Hartikainen, J. 2006. Numerical simulation of permafrost depth at Olkiluoto. Posiva Working Report 2006-52. Posiva Oy, Eurajoki, Finland.

Hedin, A. 1997. Spent nuclear fuel – how dangerous is it? SKB Technical Report TR-97-13. Swedish Nuclear Fuel and Waste Management Co (SKB), Stockholm, Sweden.

Hellä, P., Ikonen, A., Mattila, J., Torvela, T., Wikström, L. 2009. RSC-Programme – Interim Report. Approach and Basis for RSC Development, Layout Determining. Features and Preliminary Criteria for Tunnel and Deposition Hole Scale. Eurajoki, Finland. Posiva Oy. 118 p. Working Report 2009-29.

Hjerpe, T., & Broed, R. 2010. Radionuclide transport and dose assessment modelling in Biosphere assessment 2009. In preparation

Hjerpe, T., Ikonen, A.T.K., & Broed, R. 2010. Biosphere Assessment report 2009. Posiva Report POSIVA 2010-03. Posiva Oy, Eurajoki, Finland.

Hökmark, H. 2004. Hydration of the bentonite buffer in a KBS-3 repository. Appl. Clay Sci., 26, 219–233.

IAEA 2006. Geological disposal of radioactive waste – safety requirements. IAEA Safety Standards Series WS-R-4. International Atomic Energy Acency, Vienna, May 2006.

ICRP 2000. Radiation protection recommendations as applied to the disposal of long-lived solid radioactive waste. ICRP (International Commission on Radiological Protection), Publication 81. Annals of the ICRP 28(4)

ICRP 2007. Assessing Dose of the Representative Person for the Purpose of Radiation Protection of the Public and the Optimisation of Radiological Protection : Broadening the Process. ICRP (International Commission on Radiological Protection), Publication 101. Annals of the ICRP, 36, 3.

Ikonen, A.T.K. 2007. Meteorological data and update of climate statistics of Olkiluoto, 2005-2006. Posiva Working Report 2007-86. Posiva Oy, Eurajoki, Finland.

Ikonen, A.T.K., Gunia, M. & Helin, J. 2010. Terrain and ecosystem development model of Olkiluoto site, version 2009. Posiva Oy, Working Report. In preparation

Ikonen, K. 2003a. Thermal analysis of spent nuclear fuel repository. Posiva Report POSIVA 2003-04. Posiva Oy, Eurajoki, Finland.

Ikonen, K. 2003b. Thermal analysis of KBS-3H type repository. Posiva Report POSIVA 2003-11. Posiva Oy, Eurajoki, Finland.

Ikonen, K. 2005a. Mechanical analysis of cylindrical part of canisters for spent nuclear fuel. Posiva Working Report 2005-12. Posiva Oy, Eurajoki, Finland.

Ikonen, K. 2005b. Thermal Analysis of Repository for Spent EPR-type Fuel. Posiva Report POSIVA 2005-06. Posiva Oy, Eurajoki, Finland.

Internationale Zeitschrift für Kernenergie 2008. Position of the Working Group on "Scenario Development": Handling of Human Intrusion into a Repository for Radioactive Waste in Deep Geological Formations. Sonderdruck aus Internationale Zeitschrift für Kernenergie, Jaahrgang LIII (2008), Heft 8/9 August/September.

Johansson, E., Hagros, A., Autio, J. & Kirkkomäki, T. 2007. KBS-3H Layout Adaptation 2007 for the Olkiluoto Site. Posiva Working Report 2007-77. Posiva Oy, Eurajoki, Finland.

Johnson, L.H. & Tait, J.C. 1997. Release of segregated radionuclides from spent fuel. SKB Technical Report TR-97-18. Swedish Nuclear Fuel and Waste Management Co (SKB), Stockholm, Sweden.

Jones, J., Vahlund, F. & Kautsky, U. 2004. Tensit – A novel probabilistic simulation tool for safety assessments. Tests and verifications using biosphere models. Stockholm, Sweden: Swedish Nuclear Fuel and Waste Management Co. (SKB). Technical Report TR-04-07.

Karvonen, T. 2008. Surface and near-surface hydrological model of Olkiluoto Island. Posiva Working Report 2008-17. Posiva Oy, Eurajoki, Finland.

Karvonen, T. 2009a. Increasing the reliability of the Olkiluoto surface and near-surface hydrological model. Posiva Working Report 2009-07. Posiva Oy, Eurajoki, Finland.

Karvonen, T. 2009b. Development of the SVAT model for computing water and energy balance of the forest intensive monitoring plots on Olkiluoto Island. Posiva Working Report 2009-35. Posiva Oy, Eurajoki, Finland.

Karvonen, T. 2009c. Hydrological modelling in Terrain and Ecosystem Forecasts 2009 (TESM-2009). Posiva Working Report 2009-128. Posiva Oy, Eurajoki, Finland

Keto, P. & Rönnqvist, P-E. 2006. Backfilling of deposition tunnels, block alternative. Posiva Working Report 2006-89. Posiva Oy, Eurajoki, Finland.

Keto, P., Jonsson, E., Dixon, D.A., Börgesson, L., Hansen, J. & Gunnarsson, D. 2009. Assessment of backfill design for KBS-3V repository. Swedish Nuclear Fuel and Waste Management Co (SKB), Stockholm, Sweden. (To be published).

King, F. 2002. Corrosion of copper in alkaline chloride environments, SKB Technical Report TR-02-25, Swedish Nuclear Fuel and Waste Management Co (SKB), Stockholm, Sweden.

King, F., Ahonen, L., Taxén, C., Vuorinen, U. & Werme, L. 2001/2002. Copper corrosion under expected conditions in a deep geologic repository. Report POSIVA 2002-01, Posiva Oy, Olkiluoto, Finland and SKB Technical Report TR-01-23, Swedish Nuclear Fuel and Waste Management Co (SKB), Stockholm, Sweden.

King, F., Lilja, C., Puigdomonech, I., Pedersen, K., Pitkänen, P. & Vähänen, M. 2010. An update of the state-of-the-art report on the corrosion of copper under expected conditions in a deep geologic repository. Posiva Oy, Helsinki, Finland. (To be published).

Kirkkomäki, T. 2007. Design and stepwise implementation of the final repository (In Finnish with an English abstract), Posiva Working Report 2006-92. Posiva Oy, Eurajoki, Finland.

La Pointe, P. & Hermanson, J. 2002. Estimation of rock movements due to future earthquakes at four Finnish candidate repository sites. Posiva report POSIVA 2002-02. Posiva Oy, Helsinki, Finland.

Lahdenperä, A.-M., Palmén, J. & Hellä, P. 2005. Summary of overburden studies at Olkiluoto with an emphasis on geosphere-biosphere interface. Posiva Working Report 2005-11. Posiva Oy, Eurajoki, Finland.

Lanyon, G. W. & Marschall, P. 2006. Discrete fracture network modelling of a KBS-3H repository at Olkiluoto, Posiva 2006-06 and SKB report R-08-37. Posiva Oy, Eurajoki, Finland and Swedish Nuclear Fuel and Waste Management Co (SKB), Stockholm, Sweden.

Lempinen, A. 2006a. Freefem++ in THM Analyses of KBS-3V deposition hole. Posiva Working Report 2006-76. Posiva Oy, Eurajoki, Finland.

Lempinen, A. 2006b. Swelling of the buffer of KBS-3V deposition hole. Posiva Working Report 2006-77. Posiva Oy, Eurajoki, Finland.

Lempinen, A. 2006c. Simulations for EBS Task Force BMT 1. Posiva Working Report 2006-78. Posiva Oy, Eurajoki, Finland.

Lempinen, A. 2006d. THM model parameters for compacted bentonite. Posiva Working Report 2006-79. Posiva Oy, Eurajoki, Finland.

Lindberg, A. 2006. Search for glacio-isostatic faults in the vicinity of Olkiluoto. Posiva Working Report 2007-05. Posiva Oy, Eurajoki, Finland.

Liu, J. & Neretnieks, I. 1997. Coupled transport/reaction modelling with ion-exchange: study of the long-term properties of bentonite buffer in a final repository. SKI Report SKI 97:23. Swedish Nuclear Power Inspectorate (SKI), Stockholm, Sweden.

Liu, L., Moreno, L. & Neretnieks, I. 2009a. A dynamic force balance model for colloidal expansion and its DLVO-based application. Langmuir, 2009, 25(2), 679-687.

Liu, L., Moreno L. & Neretnieks, I., 2009b. A novel approach to determine the critical coagulation concentration of a colloidal dispersion with plate-like particles. Langmuir, 2009, 25(2), 688-697

Löfman J. & Poteri A. 2008, Groundwater flow and transport simulations in support of RNT-2008 analysis. Posiva Oy, Eurajoki, Finland. Posiva Working Report 2008-52. Posiva Oy, Eurajoki, Finland.

Lönnqvist, M. & Hökmark, H. 2007. Thermo-mechanical analyses of a KBS-3H deposition drift at Olkiluoto site. Posiva Working Report 2007-66. Posiva Oy, Eurajoki, Finland.

Mäntyniemi P. 2005. A tale of two earthquakes in the Gulf of Bothnia, Northern Europe in 1880s. Geophysica, 41(1-2), 73-91.

Mäntyniemi P. & Ahjos T. 1990. A catalog of Finnish earthquakes in 1610-1990. Geophysica, 26(2), 17-35.

Marcos, N., Hellä, P., Snellman, M., Nykyri, M., Pastina, B., Vähänen, M. & Koskinen, K. 2007. The role of the geosphere in Posiva's Safety Case. Paper presented at the IGSC "Geosphere Stability" workshop on the stability and buffering capacity of the geosphere for long-term isolation of radioactive waste: application to crystalline rock. Manchester, 13-15 November 2007.

Masurat, P. 2006. Potential for corrosion in disposal systems for high-level radioactive waste by *Meiothermus and Desulfovibrio*. Doctoral thesis. Göteborg University, Sweden.

Masurat, P. 2006. Potential for corrosion in disposal systems for high-level radioactive waste by *Meiothermus and Desulfovibrio*. Doctoral thesis. Göteborg University, Sweden.

Maul, P., Robinson, P., Avila, R., Broed, R., Pereira, A. & Xu, S. 2003. AMBER and Ecolego intercomparisons using calculations from SR97. Stockholm, Sweden: Swedish Nuclear Power Inspectorate (SKI) and Swedish Radiation Safety Authority (SSI). SKI 2003:28, SSI 2003:11.

Miller, B. & Marcos, N. 2007. Process report – FEPs and scenarios for a spent nuclear fuel repository at Olkiluoto. Posiva Report POSIVA 2007-12. Posiva Oy, Eurajoki, Finland.

Nagra 2002. Project Opalinus Clay – Safety Report – Demonstration of disposal feasibility for spent fuel, vitrified high-level waste and long-lived intermediate-level waste (Entsorgungsnachweis). NAGRA Technical Report 02-05. NAGRA, Wettingen, Switzerland.

NEA (Nuclear Energy Agency) 1995. Future human actions at disposal sites, safety assessment of radioactive waste repositories, a report of the NEA working group on future human actions at radioactive waste disposal sites. OECD/NEA, Paris, France.

NEA (Nuclear Energy Agency) 2000. Features, Events and Processes (FEPs) for geologic disposal of radioactive waste: an international database, OECD, Paris 2000.

NEA (Nuclear Energy Agency) 2004. Post-closure Safety Case for geological repositories. Nature and purpose. NEA No. 3679. Organisation for Economic Co-operation and Development, Nuclear Energy Agency, Paris, France.

NEA (Nuclear Energy Agency) 2007. Consideration of timescales in post-closure safety of geological disposal of radioactive waste. NEA/RWM/IGSC (2006) 3. OECD/NEA, Paris, France.

Neall, F., Pastina, B., Smith, P., Gribi, P., Snellman, M. & Johnson, L. 2007. Safety assessment for a KBS-3H spent nuclear fuel repository at Olkiluoto - Complementary evaluations of safety. Posiva Report POSIVA 2007-10. Posiva Oy, Eurajoki, Finland or SKB Report R-08-35, Swedish Nuclear Fuel and Waste Management Co (SKB), Stockholm, Sweden.

STUK 2009. Disposal of nuclear waste. Guide STUK-YVL E.5. Draft 3, 15.1.2009, in English.

Neretnieks, I. 1980. Diffusion in the rock matrix: an important factor in radionuclide migration? Journal of Geophysical Research, 85, 4379-4397.

Nilsson, K.-F., Lofaj, F., Burström, M. & Andersson, C.-G. 2005. Pressure tests of two KBS-3 canister mock-ups. SKB Technical Report TR-05-18. Swedish Nuclear Fuel and Waste Management Co (SKB), Stockholm, Sweden.

Nordman, H. & Vieno, T. 1994. Near-field model REPCOM. YJT Report YJT-94-12. Nuclear Waste Commission of Finnish Power Companies. Helsinki, Finland.

Nordman, H. & Vieno, T. 2003. Modelling of near-field transport in KBS-3V/H type of repositories with PORFLOW and REPCOM codes. Posiva Working Report 2003-07. Posiva Oy, Eurajoki, Finland.

Nykyri, M., Nordman, H., Löfman, J., Poteri, A., Marcos, N. & Hautojärvi, A. 2008. Radionuclide release and transport (RNT-2008). Posiva Report POSIVA 2008-06. Posiva Oy, Eurajoki, Finland.

Ochs, M. & Talerico, C. 2004. SR-Can. Data and uncertainty assessment. Migration parameters for the bentonite buffer in the KBS-3H concept. SKB Technical Report TR-04-18. Swedish Nuclear Fuel and Waste Management Co (SKB), Stockholm, Sweden.

Ojala, A.E.K., Virkki, H., Palmu, J-P., Hokkanen, K. & Kaija, J. 2006. Regional development of river basins in the Olkiluoto-Pyhäjärvi area, SW Finland, 2000 BP – 8000 AP. Posiva Working Report 2006-113. Posiva Oy, Eurajoki, Finland.

Pastina, B. & Hellä, P. 2006. Expected evolution of a spent nuclear fuel repository at Olkiluoto. Posiva Report POSIVA 2006-5. Posiva Oy, Eurajoki, Finland.

Påsse, T. 2001. An empirical model of glacio-isostatic movements and shore-level displacement in Fennoscandia. SKB technical report R-01-41. Swedish Nuclear Fuel and Waste Management Co (SKB), Stockholm, Sweden.

Pitkänen, P., Löfman, J., Luukkonen, A. & Partamies, S. 2003. IAEA Coordinated Research Project (CRP) - The use of selected safety indicators (concentrations, fluxes) in the assessment of radioactive waste disposal. Report 7: Site-specific natural geochemical concentrations and fluxes at four repository investigation sites in Finland for use as indicators of nuclear waste repository. Report YST-115. Geological Survey of Finland, Nuclear Waste Disposal Research.

Pohjola, J., Turunen, J. & Lipping, T. 2009. Creating high-resolution digital elevation model using thin plate spline interpolation and Monte Carlo simulation. Posiva Working Report 2009-56. Posiva Oy, Eurajoki, Finland.

Posiva, 2003. Baseline conditions at Olkiluoto. Posiva Report POSIVA 2003-02. Posiva Oy, Eurajoki, Finland.

Posiva 2005. Olkiluoto Site Description 2004. Posiva Report POSIVA 2005-03. Posiva Oy, Eurajoki, Finland.

Posiva 2008a. EIA 08: Expansion of the Repository for Spent Nuclear Fuel: Environmental Impact Assessment Programme. Posiva Oy, Eurajoki, Finland.

Posiva 2008b. Safety case plan 2008. Posiva Report POSIVA 2008-05. Posiva Oy, Eurajoki, Finland.

Posiva 2009a. TKS-2009 – Nuclear waste management at Olkiluoto and Loviisa power plants: Review of current status and future plans for 2010–2012 (in Finnish). Eurajoki, Finland: Posiva Oy.

Posiva 2009b. Olkiluoto site description 2008. Posiva Raportti POSIVA 2009-01. Posiva Oy, Eurajoki, Finland.

Pulkkanen,V-M. & Nordman, H. 2010 Modelling of near-field radionuclide transport phenomena in a KBS-3V type of repository for nuclear waste with Goldsim code – and verification against previous methods, Posiva WR 2010-14. Posiva Oy, Eurajoki, Finland.

Raiko, H. 2005. Disposal Canister for Spent Nuclear Fuel – Design Report. Posiva Report POSIVA 2005-02. Posiva Oy, Eurajoki, Finland.

Rasilainen, K. (ed.) 2004. Localisation of the SR 97 Process Report for Posiva's Spent Fuel Repository at Olkiluoto. Posiva Report POSIVA 2004-05. Posiva Oy, Eurajoki, Finland.

Robinson, P.C., Penfold, J.S.S., Little, R.H. & Walke, R.C. 2003. AMBER 4.5 verification: Summary. Document Reference Number: QRS-1059B-1. Available August 18th 2003, http://www.enviros.com/PDF/Services/Users&References_v102e .pdf.

Saanio, T., Kirkkomäki, T., Keto, P., Kukkola, T. & Raiko, H. 2006. Preliminary design of the repository - Stage 2. Posiva Working Report 2006-94. Posiva Oy, Eurajoki, Finland.

Saari, J. 2006. Local seismic network at the Olkiluoto Site - Annual Report for 2005. Posiva Working Report 2006-57. Posiva Oy, Eurajoki, Finland.

Siitari-Kauppi, M., Lukkarinen, S. & Lindberg, A. 1995. Study of rock porosity by impregnation with carbon-14-methylmethacrylate. Report YJT-95-09. Helsinki, Nuclear Waste Commission of Finnish Power Companies, Finland.

SKB 1995. SR 95 – Template for safety reports with descriptive example. SKB Technical Report 96-05. Swedish Nuclear Fuel and Waste Management Co (SKB). Stockholm, Sweden.

SKB 2006a. Long-term safety for KBS-3 repositories at Forsmark and Laxemar – a first evaluation. Main report of the SR-Can project. SKB Technical Report TR-06-09. Swedish Nuclear Fuel and Waste Management Co (SKB), Stockholm, Sweden.

SKB 2006b. Buffer and Backfill Process Report for the Safety Assessment SR-Can. SKB Technical Report TR-06-18. Swedish Nuclear Fuel and Waste Management Co (SKB), Stockholm, Sweden.

SKB 2006c. Data report for the safety assessment SR-Can. SKB Technical Report TR-06-25. Swedish Nuclear Fuel and Waste Management Co (SKB), Stockholm, Sweden.

Smith, P., Neall, F., Snellman, M., Pastina, B., Nordman, H., Johnson, L. & Hjerpe, T. 2007a. Safety assessment for a KBS-3H spent nuclear fuel repository at Olkiluoto - Summary Report. Posiva Report POSIVA 2007-06. Posiva Oy, Eurajoki, Finland or SKB Report R-08-39. Swedish Nuclear Fuel and Waste Management Co (SKB), Stockholm, Sweden.

Smith, P., Johnson, L., Snellman, M., Pastina, B. & Gribi, P. 2007b. Safety assessment for a KBS-3H spent nuclear waste repository at Olkiluoto - Evolution report. Posiva Report POSIVA 2007-08. Posiva Oy, Eurajoki, Finland or SKB Report R-08-37, Swedish Nuclear Fuel and Waste Management Co (SKB), Stockholm, Sweden.

Smith, P., Nordman, H., Pastina, B., Snellman, M., Hjerpe, T. & Johnson, L. 2007c. Safety assessment for a KBS-3H spent nuclear fuel repository at Olkiluoto - Radionuclide transport report. Posiva Report POSIVA 2007-07. Posiva Oy, Eurajoki, Finland or SKB Report R-08-38, Swedish Nuclear Fuel and Waste Management Co (SKB), Stockholm, Sweden.

Stroes-Gascoyne, S., Hamon, C. J., Kohle, C. & Dixon, D. A. 2006. The effects of dry density and porewater salinity on the physical and microbiological characteristics of highly compacted bentonite. Ontario Power Generation Report No: 06819-REP-01200-10016-R00.

STUK 2001. Long-term safety of disposal of spent nuclear fuel. Radiation and Nuclear Safety Authority (STUK). Guide YVL 8.4.

STUK 2007. Information given on the website page for environmental monitoring: http://www.stuk.fi/sateilytietoa/ulkoisen_sateilyn_valvonta/en_GB/ulk_sateily/.

Tanskanen, J. & Palmu, M. (eds.) 2004. Facility description 2003. Posiva Working Report 2004-26. Posiva Oy, Eurajoki, Finland.

Vieno, T., Peltonen, E., Vuori, S., Anttila, M., Hillebrand, K., Meling, K., Rasilainen, K., Salminen, P., Suolanen, V. & Winberg, M. 1985. Safety analysis of disposal of spent fuel – disruptive events. Report YJT-85-23 (in Finnish) Helsinki, Nuclear Waste Commission of Finnish Power Companies. Finland.

Vieno, T. & Nordman, H. 1996. Interim report on safety assessment of spent fuel disposal, TILA-96. Posiva Report POSIVA 96-17. Posiva Oy, Helsinki, Finland.

Vieno, T. & Nordman, H. 1999. Safety assessment of spent fuel disposal in Hästholmen, Kivetty, Olkiluoto and Romuvaara TILA-99. Posiva Report POSIVA 99-07. Posiva Oy, Helsinki, Finland.

Vieno, T., Hautojärvi, A., Koskinen, L. & Nordman, H. 1992. TVO-92 safety analysis of spent fuel disposal. YJT Report YJT-92-33E. Nuclear Waste Commission of Finnish Power Companies (YJT), Helsinki, Finland.

Vieno, T. & Ikonen, A.T.K. 2005. Plan for Safety Case of spent fuel repository at Olkiluoto. Posiva Report POSIVA 2005-01. Posiva Oy, Eurajoki, Finland.

Vuorela, A., Penttinen, T. & Lahdenperä, A.-M. 2009. Review of Bothnian sea shore-level displacement data and use of a GIS tool to estimate isostatic uplift. Posiva Working Report 2009-17. Posiva Oy, Eurajoki, Finland.

Werme, L.O., Johnson, L.H., Oversby, V.M., King, F., Spahiu, K., Grambow, B. & Shoesmith, D.W. 2004. Spent fuel performance under repository conditions: A model for use in SR-Can. SKB Technical Report TR-04-19. Swedish Nuclear Fuel and Waste Management Co (SKB), Stockholm, Sweden.

Wersin, P., Birgersson, M., Olsson, S., Karnland, O. & Snellman, M. 2007. Impact of corrosion-derived iron on the bentonite buffer within the KBS-3H concept. The Olkiluoto case study. Posiva Report POSIVA 2007-11. Posiva Oy, Eurajoki, Finland or SKB Report R-08-34, Swedish Nuclear Fuel and Waste Management Co (SKB), Stockholm, Sweden.

APPENDIX A: NEAR-FIELD AND GEOSPHERE MODEL ASSUMPTIONS AND THEIR CLASSIFICATION

The assumptions and model simplifications underlying the near-field, geosphere and biosphere modelling processes are summarised in tabular form in this appendix. The assumptions and simplifications have been classified according to the scheme shown in Table A-1, which is a modified version of the approach presented by Swiss National Cooperative for the Disposal of Radioactive Waste (Nagra 2002). The classification of specific near-field and geosphere transport modelling assumptions and simplifications is shown in Tables A-2 to A-5. Similar tables for biosphere assessment modelling are given in Broed et al. (2007) and (Hjerpe et al. 2010). It should be stressed that these tables focus only on key model assumptions. A more systematic description of the models and their underlying assumptions will be given in the Models and Data Report.

Table A-1. *Classification of conceptual assumptions and simplifications.*

Categorisation of assumptions for the broad characteristics and evolutionary path followed by the near-field and geosphere and conceptualisation of phenomena	
C-LE	Conceptual assumption corresponds to the likely/expected characteristics and evolution of the system
C-PCA	Pessimistic conceptual assumption within the reasonably expected range of possibilities
WRP	Within the range of possibilities, but likelihood not currently possible to evaluate - other (and sometimes more pessimistic) assumptions may not be unreasonable
C-ST	Stylised conceptualisation of system characteristics and evolution
Categorisation of simplifications made for modelling purposes	
M-MS	Model simplification - not significantly affecting numerical results (as shown by scoping calculations or more qualitative arguments)
M-CS	Model simplification - intrinsically conservative
M-CP	Model simplification - conservative given the model parameters that are also assumed

Table A-2. *Assessment of general assumptions applying to all repository assessment scenarios.*

Assumption	Class	Comment
The deep underground location of the repository is maintained over several million years, isolating the spent fuel from the surface environment.	C-LE	Eventually, after hundreds to thousands of millions of years, slow geological processes could result in the rock being gradually eroded, so that repository materials eventually reach the surface (Section 9.3).
Radioactive decay and ingrowth are described by the Bateman equations.	C-LE	The Bateman equations give the number of atoms of each nuclide of a radioactive decay chain produced after a specified time, when a specified number of atoms of the parent nuclide are initially present.
The canister remains located coaxially with the deposition hole and no significant canister sinking takes place over the assessment period.	C-LE	No processes have been identified that could reduce the density of the buffer to such an extent that canister sinking is possible.
Criticality does not occur.	C-LE	Anttila (2005) has shown that BWR and VVER fuel remains subcritical even if the void inside the canister is entirely filled with water. Burnup credit needs to be used in the case of EPR/PWR fuel. Further studies are planned concerning criticality scenarios during the long-term evolution of the canister.

Table A-3. *Assessment of assumptions regarding canister failure and radionuclide release from failed canisters.*

Assumption	Class	Comment
Canister failure can occur by one of three modes: (i) initial defect - penetrating or non-penetrating (ii), enhanced corrosion due to perturbed buffer and (iii), rock shear.	C-LE	A fourth mode - failure due to isostatic loading - is ruled out (Section 6.3.3). For future safety assessments, the possibility of an initial penetrating defect may be also ruled out or may be retained as a stylised conceptualisation of canister failure (ST).
In the majority of calculation cases, the failed canister is assumed to contain BWR fuel from Olkiluoto 1-2.	WRP	Alternative calculation cases consider other fuel types; the impact of fuel type is shown to be small.

Once a canister fails water builds up inside.	M-CS	The consumption of water by corrosion of the insert is the most likely outcome in the case of a small penetrating defect (Pastina & Hellä 2006, Gribi et al. 2007).
Once a canister fails, water contacts the fuel immediately. Delay, due e.g. to the intactness of many of the cladding tubes and to the generation of gas inside the canisters, is neglected.	M-CS	Assumption applies to the 2009 safety analysis of a KBS-3V repository. A delay is assumed in the KBS-3H safety assessment.
The transport resistance of a penetrating defect remains constant over the assessment period.	C-ST	Assumption applies to most calculation cases in KBS-3V safety analysis. Complete loss of transport resistance is assumed in the KBS-3H safety assessment after a specified time.
Radionuclides that are concentrated at grain boundaries in the fuel, at pellet cracks and in the fuel / cladding gap are released instantaneously upon contact with water entering a failed canister.	M-CS	Though the assumption of instant release is conservative, the proportion of the overall inventory of a radionuclide that is assumed to be instantly released (the IRF) is subject to uncertainty.
Radionuclides in the fuel matrix are released congruently as the fuel dissolves.	C-LE	Experimental evidence supports the assumption of congruent release (Gray 1999; Röllin et al. 2001).
A constant rate of fuel dissolution is assumed over the assessment period.	C-LE	The rate at which dissolution occurs is determined primarily by redox conditions in the immediate vicinity of the fuel surfaces. Redox conditions are expected to be reducing due to the presence of hydrogen and corroding iron (see, e.g. Carbol et al. 2005).
Radionuclides in the fuel cladding and in other metal parts are released congruently as these components corrode.	C-LE	
A constant corrosion rate is assumed for the fuel cladding and for other metal parts, until the corrosion process is complete.	C-LE	The fractional corrosion rate of Zircaloy cladding (10^{-4} per year) and that of other metal parts (10^{-3} per year) are pessimistic.
Dissolved radionuclides are assumed to be uniformly mixed in the void space in the canister interior (the transport resistance provided by	M-CS	

constricted internal spaces - e.g. the fractured cladding - that could lead to non-uniformity is neglected) and sorption on corrosion products inside the canisters is not considered.		
Solubility limits constrain the aqueous concentrations of certain radionuclides within the canisters, with precipitation occurring if the solubility limits of the corresponding elements are exceeded, and redissolution occurring if concentrations fall.	C-LE	
Any radionuclide-bearing colloids formed when solubility limits are exceeded are retained within the canister interior.	C-LE	The buffer is expected to provide a colloid filter.
C is released by the corrosion of activated metal parts is in gaseous form (methane).	M-CS	Conservative given the assumption that methane does not sorb in either the near field or geosphere. There are uncertainties in the speciation of C-14. According to the few studies available (e.g. Van Konynenburg 1994), C-14 is released mainly from metal parts or Zircaloy as carbide, which in water may form organic compounds e.g., short chain carboxylic acids, alcohols and aldehydes, or methane (in presence of hydrogen).
Only the concentrations of isotopes originating from the spent fuel are taken into account in evaluating whether solubility limits are exceeded in the canister interiors; the background concentrations of isotopes originating elsewhere are conservatively ignored.	M-CS	
Immobilisation by co-precipitation with secondary minerals derived from the fuel and canister corrosion is neglected.	M-CS	

Table A-4. Assessment of assumptions regarding radionuclide transport through the repository near field to the bedrock.

Assumption	Class	Comment
Radionuclide transport in the buffer occurs only by aqueous diffusion (with the exception of cases considering the impact of repository-generated gas).	M-MS	Other processes are either excluded (e.g. colloid-facilitated transport) or, in the case of advection, are shown to be so small as to be negligible.
Diffusion is well described by Fick's laws.	C-LE	
Some radionuclides are subject to anion exclusion in the buffer, affecting their diffusion coefficients and the effective porosity that they encounter.	C-LE	Buffer pore surfaces, being negatively charged, repel anions. Anion concentrations in narrow pores and near to pore surfaces in larger pores are therefore less than in the case of neutral and cationic species, for given concentrations at the boundaries.
Transport in the buffer is retarded by linear, equilibrium sorption.	M-CP	Sorption may, in reality, be non-linear, but K_d values can be selected to handle non-linearity conservatively.
Solubility limits based on (average) buffer porewater composition constrain radionuclide concentrations at the buffer/rock interface; any radionuclide-bearing colloids are immobile.	M-MS	In reality, solubility limits constrain radionuclide concentrations throughout the buffer. Near the interface with the bedrock, the buffer porewater composition will be intermediate between groundwater composition and porewater composition in the bulk of the buffer.
Solubility limits and sorption coefficients in the repository near field are based on a pore water composition that is derived assuming a dilute, brackish groundwater in the majority of calculation cases.	C-ST	Groundwater composition will vary over time, in particular in response to current land uplift and the impact of future major climate changes (glaciation). Alternative calculation cases with other groundwater compositions are considered.
Only the concentrations of isotopes originating from the spent fuel are taken into account in evaluating whether solubility limits are exceeded at the buffer/rock interface.	M-CS	The background concentrations of isotopes originating, for example, from the groundwater are conservatively ignored.
The transport-relevant properties of the buffer and backfill are constant in space and time, for all times following the start of radionuclide release from the canister.	M-MS	The evolving saturating state will result in evolving transport properties, but the effects on release from the buffer are expected to be small.

In the majority of calculation cases, repository-generated gas has no impact on radionuclide transport.	WRP	Alternative calculation cases address the possibilities of (i), expulsion of contaminated water by gas, and (ii), transport of C-14 in volatile form with repository gas.
In the majority of calculation cases, the transport properties of the buffer / rock interface are not significantly perturbed by phenomena such as thermally-induced rock spalling.	WRP	"What-if", "sensitivity" and/or "supplementary" calculation cases consider a perturbed buffer/rock interface.
Radionuclides are transferred from the buffer to a host rock fracture intersecting a deposition hole or the deposition drift at the point closest to the location of canister failure.	C-PCA	Other release paths are also considered in the case of a KBS-3V repository: (i) from the buffer to the backfill in the upper part of the deposition hole and hence to the deposition tunnel EDZ; and (ii), from the buffer to the deposition tunnel backfill and thence to its EDZ.

Table A-5. *Assessment of assumptions regarding radionuclide transport in the bedrock.*

Assumption	**Class**	**Comment**
Advective transport through the geosphere is characterised by a transport resistance, the value of which is assigned conservatively based on the results of groundwater flow modelling.	M-CP	Uncertainties in transport resistance arising from uncertainties in groundwater flow modelling and in the location of the failed canister(s) within the bedrock are taken into account by considering a range of transport resistance values in "sensitivity" calculation cases.
Spreading of radionuclide releases by longitudinal dispersion along the path is neglected.	M-CP	In reality, radionuclides will migrate from a failed canister along a heterogeneous set of transport paths. Longitudinal dispersion may arise due, for example, to the variability in transport times between one path and another. The omission of longitudinal dispersion is conservative with respect to the magnitude of activity flux from the geosphere to the biosphere provided the transport resistance of the geosphere is selected conservatively, taking geosphere heterogeneity into account.
Matrix diffusion is well described by Fick's laws.	C-LE	
Matrix diffusion occurs only within a limited volume of rock adjacent to	M-CP	The connected rock matrix porosity can generally only be demonstrated to exist within a few centimetres of fracture

the bedrock fractures		surfaces. Diffusion into stagnant water pools in the fractures is conservatively omitted.
Some radionuclides are subject to anion exclusion in the rock matrix, affecting their diffusion coefficients and the effective porosity that they encounter.	C-LE	Rock-matrix pore surfaces, being negatively charged, repel anions.
Transport in the rock matrix is retarded by linear, equilibrium sorption.	M-CP	Sorption may, in reality, be non-linear, but K_d values can be selected to handle non-linearity conservatively.
Sorption occurs only on rock matrix pore surfaces.	M-CS	Sorption on fracture fillings is conservatively omitted.
Sorption coefficients are based on the assumption of dilute, brackish groundwater.	C-ST	Groundwater composition will vary over time, in particular in response to current land uplift and the impact of future major climate changes (glaciation). "Sensitivity" calculation cases with other groundwater compositions are undertaken.
Colloid-facilitated radionuclide transport is negligible.	C-LE	Colloid concentrations are low in Olkiluoto groundwater. Colloids may also form in the repository near field, e.g. by precipitation, but they are generally not able to pass through the bentonite buffer on account of its fine pore structure (unless the buffer is eroded or undergoes mineralogical changes that lead to embrittlement and fracturing).
A gas phase is not naturally present in the bedrock, and repository-generated gas has no significant impact on groundwater flow or geosphere transport at times when radionuclide releases to the geosphere may occur.	C-LE	At the Olkiluoto site there is naturally occurring gas dissolved in the groundwater, but no signs of meaningful amounts of gas in the gas phase have been found. H_2 generation from corrosion of the KBS-3H steel supercontainer shell will be significant, but will have largely ceased by the time most radionuclides are released from failed canisters, except possibly in the tightest sections of rock where groundwater flow is in any case virtually zero. H_2 generation from the cast iron insert will, however, require further consideration in future studies.
No account taken of irreversible sorption or long-term immobilisation processes (precipitation/co-precipitation) in the host rock.	C-ST	Conservative, unless a change in geochemical conditions leads to remobilisation of previously immobilised radionuclides.

The sealed access tunnel system, exploration boreholes and their surrounding excavation disturbed zones do not provide preferential paths for radionuclide transport.	C-LE	The impact of these features will be considered in future groundwater flow modelling studies.

REFERENCES TO APPENDIX A

Anttila M. 2005. Radioactive characteristics of the spent nuclear fuel of the Finnish nuclear power plants. Posiva Working Report 2005-71. Posiva Oy, Eurajoki, Finland.

Broed, R., Avila, R., Bergström, U., Hjerpe, T. & Ikonen, A. 2007. Biosphere analysis for selected cases in TILA-99 and in KBS-3H safety evaluation, 2007. Posiva Working Report 2007-109. Posiva Oy, Eurajoki, Finland.

Carbol, P., Cobos-Sabate, J., Glatz, J-P., Ronchi, C., Rondinella, V., Wegen, D.H., Wiss, T., Loida, A., Metz, V., Kienzler, B., Spahiu, K., Grambow, B., Quiñones, J. & Valiente, A.M.E. 2005. The effect of dissolved hydrogen on the dissolution of ^{233}U doped $UO_2(s)$, high burn-up spent fuel and MOX fuel. Stockholm, Sweden: Swedish Nuclear Fuel and Waste Management Co. (SKB). Technical Report TR-05-09.

Gray, W. J. 1999. Inventories of iodine-129 and cesium-137 in the gaps and grain boundaries of LWR spent fuels. Mat. Res. Soc. Symp. Proc. Vol. 556, 487-494.

Gribi, P., Johnson, P., Suter, D., Smith, P., Pastina, B. & Snellman, M. 2007. Safety assessment for a KBS-3H spent nuclear fuel repository at Olkiluoto - Process Report. Posiva Report POSIVA 2007-09. Posiva Oy, Eurajoki, Finland or SKB Report R-08-36. Swedish Nuclear Fuel and Waste Management Co (SKB), Stockholm, Sweden.

Hjerpe, T., Ikonen, A.T.K., & Broed, R. 2010. Biosphere Assessment report 2009. Posiva Report POSIVA 2010-03. Posiva Oy, Eurajoki, Finland.

Nagra 2002. Project Opalinus Clay – Safety Report – Demonstration of disposal feasibility for spent fuel, vitrified high-level waste and long-lived intermediate-level waste (Entsorgungsnachweis). Nagra Technical Report 02-05. Nagra, Wettingen, Switzerland.

Pastina, B. & Hellä, P. 2006. Expected Evolution of a Spent Nuclear Fuel Repository at Olkiluoto. Posiva Report POSIVA 2006-5. Posiva Oy, Eurajoki, Finland.

Röllin S., Spahiu K. & Eklund U. B. 2001. Determination of dissolution rates of spent fuel in carbonate solutions under different redox conditions with a flow- through experiment. J. Nucl. Mater. 297 (2001) 231–243.

Van Konynenburg, R.A. 1994. Behaviour of carbon-14 in waste packages for spent fuel in a tuff repository. Waste Management, 14, 363-383.

LIST OF REPORTS

POSIVA-REPORTS 2010

POSIVA 2010-01	Models and Data Report 2010 *Barbara Pastina*, Saanio & Riekkola Oy *Pirjo Hellä*, Pöyry Oyj ISBN 978-951-652-172-8 March 2010
POSIVA 2010-02	Interim Summary Report of the Safety Case 2009 Posiva Oy ISBN 978-951-652-173-5 March 2010