POSIVA 2008-05

Safety Case Plan 2008

Posiva Oy

July 2008

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SAFETY CASE PLAN 2008

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 the construction license application for a spent fuel repository by the end of the year 2012. The long-term safety section supporting the license application is based on a safety case, which, according to the internationally adopted definition, is 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. In 2005, Posiva presented a plan to prepare such a safety case. The present report provides a revised plan of the safety case contents mentioned above.

The update of the safety case plan takes into account the recommendations made by the Radiation and Nuclear Safety Authority (STUK) about improving the focus and further developing the plan. Accordingly, particular attention is given to the quality management of the safety case work, the management of uncertainties and the scenario methodology. The quality management is based on the ISO 9001:2000 standard process thinking enhanced with special features arising from STUK's YVL Guides.

The safety case production process is divided into four main sub-processes. The *conceptualisation & methodology* sub-process defines the framework for the assessment. The *critical data handling and modelling* sub-process links Posiva's main technical and scientific activities to the production of the safety case. The *assessment* sub-process analyses the consequences of the evolution of the disposal system in various scenarios, classified either as part of the expected evolution or as disruptive scenarios. The *compliance & confidence* sub-process is responsible for final evaluation of compliance of the assessment results with the regulatory criteria and the overall confidence in the safety case.

As in the previous safety case plan, the safety case will be based on several reports, but changes are introduced to the contents of the report ”portfolio”. The new portfolio includes the following reports: *Description of the Disposal System, Process, Formulation of Scenarios* (including repository evolution), *Models and Data, Analyses of Scenarios, Complementary Considerations and Summary.* A key novelty is the *Models and Data Report*, which links the safety case work to the site characterisation, and to the programme for the repository's technical design and development. The purpose of the report is to improve the data traceability and the transparency of the theoretical basis, and to describe and organise the quality management of data production in an efficient way. The first *Models and Data Report* will be published in 2009.

Avainsanat - Keywords

safety case, safety assessment, long-term safety, spent fuel, nuclear waste, crystalline bedrock

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SUUNNITELMA TURVALLISUUSPERUSTELUSTA 2008

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 (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. Posiva on vuonna 2005 esittänyt suunnitelman tällaisen turvallisuusperustelun laatimisesta. Käsillä olevassa raportissa esitetään tarkennettu ja päivitetty suunnitelma tuvallisuusperustelun muodostavasta aineistosta ja sen laatimisesta.

Suunnitelman päivityksessä on huomioitu Säteilyturvakeskuksen (STUK) eri yhteyksissä esittämät suositukset suunnitelman tarkentamisesta ja kehittämisestä. Näiden mukaisesti huomiota on kiinnitetty erityisesti turvallisuusperustelun tuottamiseen liittyvän työn laadunhallintaan, epävarmuuksien hallintaan ja skenaarioanalyysiin. Laadunhallinta perustuu uusitussa suunnitelmassa ISO 9001:2000 -standardin mukaiseen prosessiajatteluun mutta samalla on otettu huomioon ydinlaitosten johtamisjärjestelmää koskevat viranomaisvaatimukset.

Turvallisuusperustelun tuottamisprosessi on jaettu neljään alaprosessiin. *Konseptualisointi ja metodologia* -prosessin avulla määritetään koko turvallisuusanalyysin viitekehys. *Kriittisten lähtötietojen käsittelyyn ja mallintamiseen* liittyvä prosessi toimii keskeisenä yhdyssiteenä Posivan teknis-tieteellisen toiminnan ja turvallisuusperustelun välillä. *Turvallisuusanalyysi* prosessi analysoi loppusijoitus-järjestelmän kehityskulkujen seuraukset sekä odotettavissa olevien että poikkeavien kehityskulkujen avulla. *Turvallisuusvaatimusten täyttymisen ja luotettavuuden osoittaminen* (*Compliance and confidence)* – prosessin tehtävänä on arvioida viranomaisvaatimusten täyttyminen analyysin tulosten perusteella ja turvallisuusperustelun luotettavuutta kokonaisuudessaan.

Turvallisuusperustelu tulee edelleenkin perustumaan useaan erilliseen raporttiin mutta "porfolion" sisältöön on tehty muutoksia. Uusi raporttisalkku muodostuu seuraavista raporteista: *Loppusijoitusjärjestelmä, Prosessiraportti*, *Skenaarioiden muodostaminen* (sisältäen kuvauksen sijoituspaikan ja loppusijoitustilan kehityksestä), *Mallit ja lähtötiedot*, *Skenaarioanalyysit*, *Täydentävät turvallisuustarkastelut* sekä *Yhteenvetoraportti*). Merkittävin lisäys on malleja ja lähtötietoja koskeva raportti (*Models and Data Report*), joka yhdistää turvallisuusperustelutyön, paikkatutkimusten ja teknistä loppusijoitusjärjestelmää koskevan suunnittelu- ja kehitystyön. Raportin on tarkoitus parantaa entisestään lähtötietojen jäljitettävyyttä ja teoreettisten perustelujen läpinäkyvyyttä siten, että samalla turvallisuuden kannalta kriittisten lähtötietojen tuottamisen laadunhallinta voidaan tehokkaasti kuvata ja organisoida. Uudesta malleja ja dataa käsittelevästä raportista ensimmäinen versio ilmestyy vuoden 2009 lopulla.

Avainsanat – Keywords

Turvallisuustodisteet, turvallisuusanalyysi, pitkäaikaisturvallisuus, käytetty ydinpolttoaine, loppusijoitus, kiteinen kallioperä

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1 INTRODUCTION

1.1 Background and scope of this report

A safety case is 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 2004a, Figure 1-1). A safety case includes a quantitative safety assessment, which is defined as the process of systematically analysing the ability of the disposal facility to provide the safety functions and to meet technical requirements, and to evaluate the potential radiological hazards and compliance with the safety requirements. The safety case broadens the scope of the safety assessment to include the compilation of a wide range of evidence

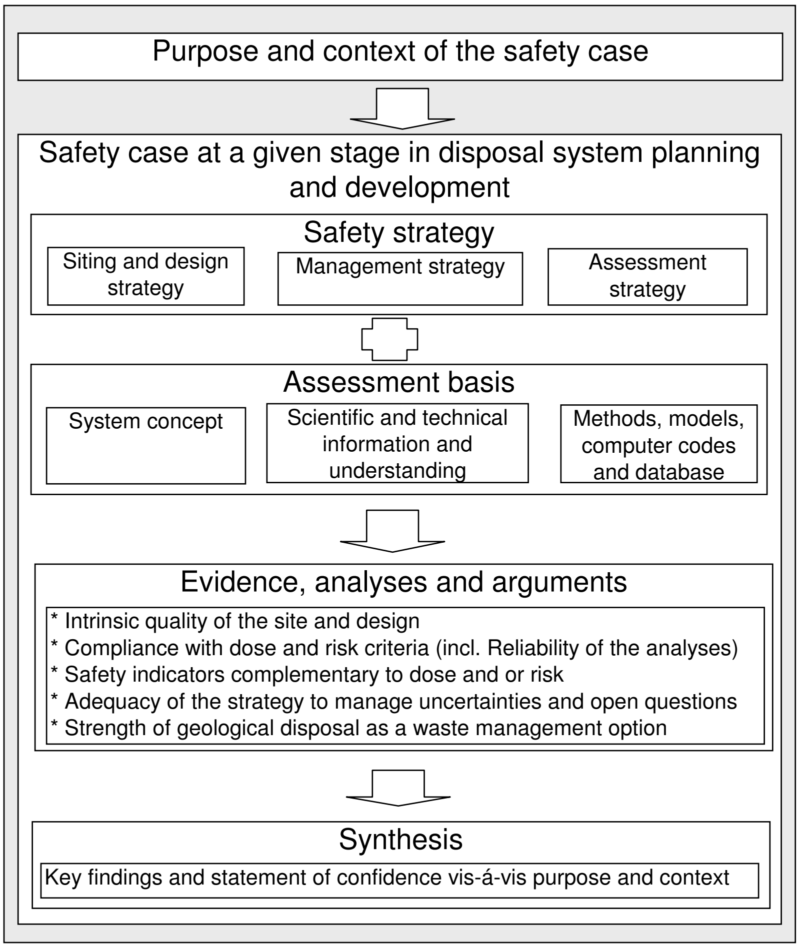

Figure 1-1. *NEA's overview of the elements of a safety case and their relationships (NEA 2004a, Figure 1).*

and arguments that complement and support the reliability of the results of the quantitative analyses. In concrete terms, a safety case includes all material presented by the repository implementer to the authorities and to other stakeholders in support of an application to site, construct, operate or close a disposal facility. The safety case is a key input to decision-making at several steps in the repository planning and implementation process. It becomes more comprehensive and rigorous as the programme progresses. The present report provides a plan for the production of a safety case to support a future construction license application for a KBS-3 repository at the Olkiluoto site in Finland. This replaces the earlier Safety Case Plan published by Posiva in 2005 (Vieno & Ikonen 2005).

The KBS-3 concept for spent fuel disposal in crystalline bedrock was first introduced by the Swedish SKB in 1983 and has since been subject to intense research and development work both in Sweden and in Finland (SKBF/KBS 1983). The KBS-3 concept, which is based on the multiple barrier principle, aims at long-term isolation and containment of spent fuel assemblies within durable copper-iron canisters in a way that any releases of radionuclides from the canisters are prevented for as long as they could cause any harm to man or the environment. The repository design allows for retrieval of the spent fuel, if considered necessary. The safety concept and the safety functions of the barriers of the disposal system are discussed in Section 1.3 and Section 2.1.2, respectively.

Two variants of the KBS-3 concept are being considered. The reference concept is KBS-3V (Figure 1-2, left), in which the disposal canisters are emplaced vertically in individual deposition holes. The alternative concept is KBS-3H (Figure 1-2, right), in which the canisters are emplaced horizontally in long deposition drifts. Since 2002, the feasibility of the KBS-3H concept has been studied by SKB and Posiva in a joint project. The results of the KBS-3H feasibility study will be published in 2008.

Several long-term safety assessments have been prepared in both Sweden and Finland based on the KBS-3 concept. In Finland, the Environmental Impact Assessment for a spent fuel disposal facility and the Decision-in-Principle (DiP) in 2001 to site such a facility at Olkiluoto were based on the TILA-99 safety assessment of a KBS-3V repository (Vieno & Nordman 1999). Subsequent safety assessment work has continued according to the Safety Case Plan published in 2005 (Vieno & Ikonen 2005). The plan was based on the idea of a portfolio of reports which together would produce the main argumentation for the long-term safety of a spent fuel repository at Olkiluoto. The portfolio structure of the 2005 Safety Case Plan is shown in Figure 1-3.

The geoscientific basis of the safety case for both the KBS-3V and KBS-3H concepts has thus far been based on the information presented in Olkiluoto Site Description reports (Posiva 2003a; 2005; Andersson *et al.* 2007). These reports describe the present situation at, and past evolution of, the Olkiluoto site, and disturbances caused by the underground characterisation facility, ONKALO, which is currently being constructed and will later also serve as an access route to the repository. Similarly, a large number

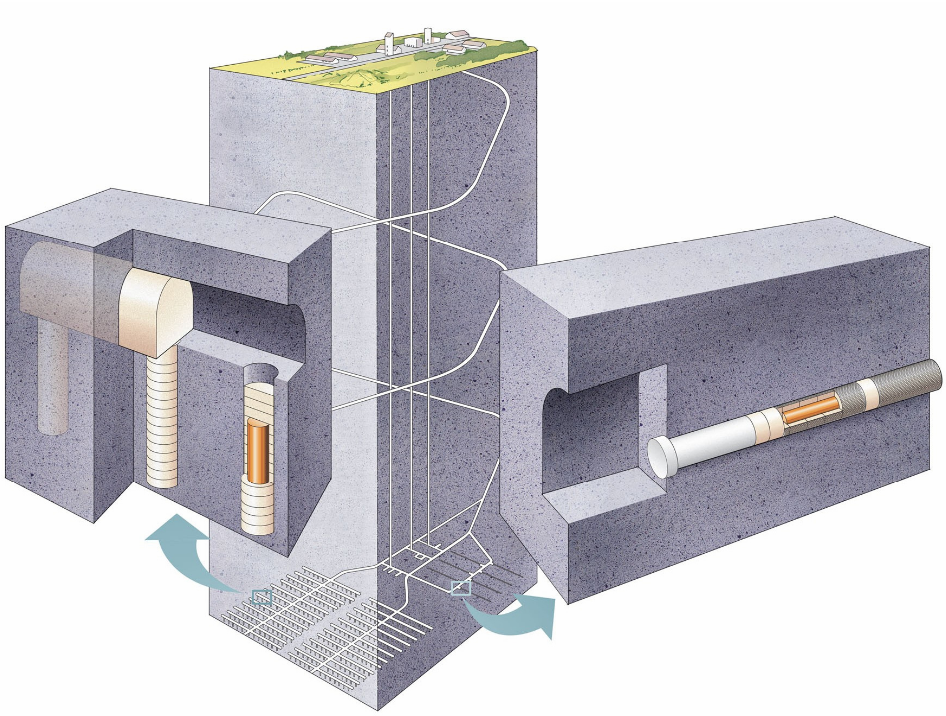

Figure 1-2. *The KBS-3V (left) and KBS-3H (right) variants of the KBS-3 spent fuel disposal concept.*

of reports have been published on work related to the engineering barrier system (EBS) of a KBS-3 repository. A summary of the development work is included in Posiva´s latest three-year programme for Research, Development and Technical Design TKS-2006 (Posiva 2006).

The scientific understanding supporting the KBS-3V safety studies is synthesised in a **Process Report,** an **Evolution Report,** and in a **Biosphere Assessment Report**. The Process Report, as its name indicates, describes thermal, hydraulic, mechanical and chemical (THMC) processes affecting repository evolution (e.g. thermal pulse, resaturation, swelling of the buffer), including microbiological as well as gas and radiation-related processes, and discusses their long-term safety relevance.

The Evolution Report describes how the main processes identified in the Process Report affect the expected evolution of a repository at Olkiluoto, and describes this evolution in a broadly chronological manner, highlighting the interactions between the processes. Updated versions of the Process Report (Miller & Marcos 2007) and of the Evolution Report (Pastina & Hellä 2006) have recently been published, discussing the KBS-3V concept.

The Biosphere Assessment Report summarises the outcome of the biosphere assessment, which describes the past, present and future conditions of the surface system of the Olkiluoto site, tracks the fate of hypothetical releases of radionuclides from the repository reaching the biosphere, and assesses possible radiological consequences to humans and other biota.

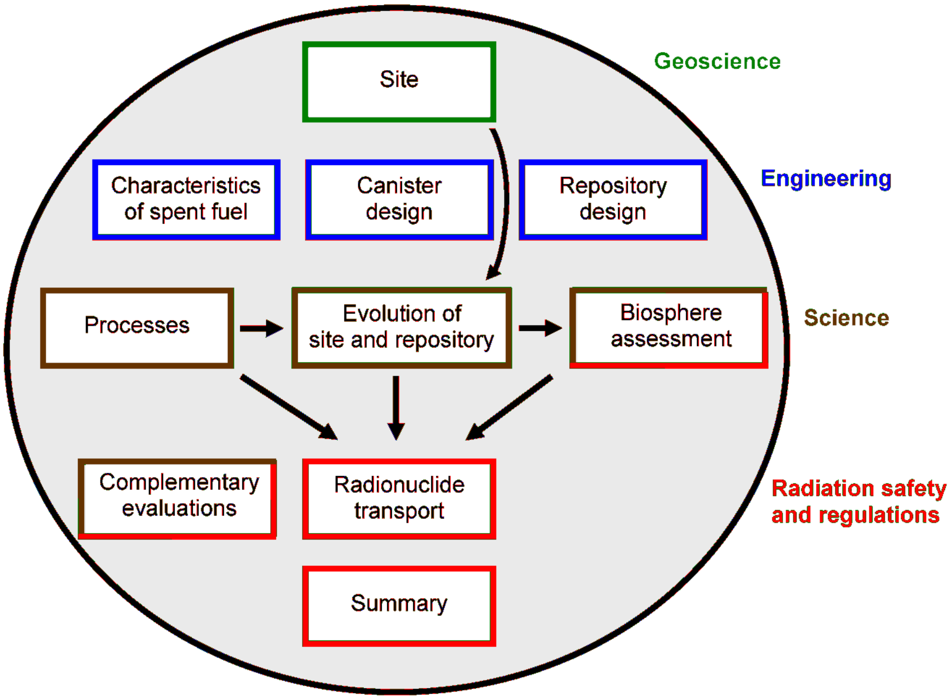

Figure 1-3. *The main reports in the safety case portfolio as described in the 2005 Safety Case Plan by Vieno & Ikonen (2005). The colours of the boxes indicate the domain of the reports (geosciences, engineering, science or radiation safety and regulations) and the arrows show the most important transfers of knowledge and data.*

The Process and Evolution Reports provide the basis for the selection of the calculation cases whereby the consequences of radionuclide releases in the event of canister failure are evaluated, as described in the **Radionuclide Transport Report** and the **Biosphere Assessment Report**. A Radionuclide Release and Transport Report for a KBS-3V repository, which provides an update of the analyses reported in TILA-99, will be published in mid-2008 (Nykyri *et al.* 2008). The corresponding Biosphere Assessment Report will be published in 2009.

For the KBS-3H design variant, a safety assessment based on the Olkiluoto site has been jointly compiled by Posiva and SKB. The methodology for the safety assessment is based broadly on that used in the recent Swedish safety assessment SR-Can (SKB 2006a), but the presentation of the main components of the assessment follows the portfolio structure of the 2005 Safety Case Plan and gives a good example of its integral application. The purpose of the KBS-3H safety assessment has been to support the feasibility assessment of the KBS-3H concept. It therefore focuses mainly on those aspects that are judged to have a different significance to, or potential impact on, KBS-3H compared with KBS-3V.

Posiva has published the following reports jointly with SKB, documenting the KBS-3H safety assessment:

- Process Report (Gribi *et al.* 2007)

- Evolution Report (Smith *et al.* 2007a)
- Radionuclide Transport Report (Smith *et al.* 2007b)
- Complementary Evaluations of Safety Report (Neall *et al.* 2007)
- Safety Assessment Summary Report (Smith *et al.* 2007c).

A Biosphere Analysis Report (Broed *et al.* 2007) was produced in parallel with the above-mentioned reports, using input from the KBS-3H Radionuclide Transport Report. The main conclusions on the ensuing releases to the biosphere are summarised in the Summary Report of the KBS-3H safety assessment.

It was concluded that KBS-3H is a promising alternative at a site with the broad characteristics of Olkiluoto from the long-term safety point of view. Posiva and SKB jointly decided to continue the development of the KBS-3H concept until at least 2010, but the vertical emplacement concept, KBS-3V, is still considered the reference design for licensing.

The safety case production, as outlined in the 2005 Safety Case Plan by (Vieno & Ikonen 2005), was organised as a project that was structured according to the main portfolio reports (Figure 1-3) and scheduled according to the publication of these reports. The schedule was later revised, but mainly retained the original idea of producing preliminary versions of the main reports before the final licensing documents.

This approach to safety case production has provided a good basis for working efficiently through a number of fairly independent tasks. However, in the review of Posiva's TKS programme (TKS-2006, Posiva 2006), STUK questioned, in particular, Posiva´s approach for the:

- overall integration of the safety case work,
- management of uncertainties,
- scenario selection, and
- management of quality.

STUK also recommended that Posiva update the 2005 Safety Case Plan. The revised plan for the safety case to support the construction license application is the subject of the present report. In light of the issues raised by STUK, a more process-oriented approach is being adopted. In this framework, the interactions among Posiva's different activities are explicitly identified and can be adopted in the practical organisation of the work. Furthermore, the need for iterations between requirements and technical options and constraints can be taken into account more efficiently.

1.2 Regulatory context

The regulatory requirements for long-term safety are set forth in the Government Decision on the safety of the disposal of spent nuclear fuel (STUK 1999) and, in more detail, in the regulatory Guide YVL 8.4 issued by STUK (2001).

The regulations emphasise the goal of long-term containment by stating that the repository design shall effectively hinder the release of disposed radioactive substances into the host rock for several thousand years.

In Guide YVL 8.4, Section 2.2, for quantitative safety assessment calculations, the regulatory requirements for a period that shall extend to at least several thousand years after the closure of the repository are stated as follows:

- *"The annual effective dose to the most exposed members of the public which shall remain below 0.1 mSv.*
- *The average annual effective doses to other members of the public which shall remain insignificantly low. The acceptability of these doses depends on the number of exposed people, but they shall not be more than 1/100 to 1/10 of the constraint for the most exposed individuals, i.e. no more than 0.001 to 0.01 mSv per year".*

Section 2.5 of Guide YVL 8.4 gives the following guidance on the protection of other living nature that shall be considered in the safety assessment.

"Disposal of spent fuel 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. These 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. Moreover, rare animals and plants as well as domestic animals shall not be exposed detrimentally as individuals".

In the long term, after several thousand years, the quantitative regulatory criteria are based on constraints on the release rates of long-lived radionuclides from the geosphere to the biosphere. The nuclide-specific constraints are set out in Guide YVL 8.4. The constraints apply to activity releases which arise from the expected evolution scenarios. Concerning the assessment for these scenarios, according to Guide YVL 8.4, Section 2.3:

"*the activity releases can be averaged over 1000 years at the most. The sum of the ratios between the nuclide-specific activity releases and the respective constraints shall be less than one".*

Section 2.4 of Guide YVL 8.4 also discusses the treatment of unlikely disruptive events:

"The importance to long-term safety of unlikely disruptive events shall be assessed and, whenever practicable, the acceptability of the consequences and expectancies of radiation impacts caused by such events shall be evaluated in relation to the dose and release rate constraints".

The unlikely disruptive events considered shall include at least

- *"boring a deep water well at the disposal site*
- *core-drilling hitting a waste canister*

- *a substantial rock movement occurring in the environs of the repository".*

Guidance on the use of models and data in safety assessment is given in Section 4.3 of Guide YVL 8.4:

"The computational methods i.e. models and data employed in the safety assessment shall be selected on the basis that the results, with high degree of certainty, overestimate the radiation exposure and radioactive release likely to occur. Simplification of the models as well as the determination of input data for them shall be based on the principle that the performance of any barrier will not be overestimated but neither overly underestimated. The various models and input data shall be mutually consistent, apart from cases where just the simplifications in modelling or the aim of avoiding the overestimation of the performance of barriers implies apparent inconsistency".

In the very long term, after several hundred thousand years (Ruokola 2002), no rigorous quantitative safety assessment is required, but the judgement of safety can be based on more qualitative considerations or "*complementary considerations, such as bounding analyses with simplified methods, comparisons with natural analogues and observations of the geological history of the site*" (Guide YVL 8.4, Section 4.4).

The safety regulations imply that the main emphasis of the safety case shall be on the isolation and containment capacity of the disposal system. For the quantitative safety assessment, Posiva´s interpretation of these requirements led to the identification of three time frames as follows:

- From emplacement to several thousand years thereafter, when the dose rate constraints apply. Biosphere transport and assessments of human exposure need to be performed only for those radionuclides that are released to the biosphere during this period.
- From several thousand to several hundred thousand years after closure, when the constraints on the activity release rates from the geosphere to the biosphere apply.
- The period after several hundred thousand years, when the safety assessment can increasingly be based on qualitative considerations.

The safety regulations are currently under revision. The general safety requirements set by the Government in 1999 will be replaced by a Government Decree and, subsequently, the relevant YVL Guide by STUK will also be revised. However, according to the information currently available, there will not be major changes in the primary criteria and requirements for long-term safety.

1.3 Posiva's safety concept

Posiva's concept of spent fuel disposal is based on

- the KBS-3 design of the geologic repository and

- the characteristics of the Olkiluoto site.

Posiva's safety concept aims at making the best possible use of these two elements to attain a level of long-term safety that is as high as reasonably achievable. Posiva’s safety concept is depicted in Figure 1-4. According to this concept, safety rests first and foremost on the ***long-term isolation and containment*** of radionuclides within the copper-iron canisters. A clay buffer protects the canisters from rock movements and potential detrimental substances, limits groundwater flow around the canisters and limits and retards radionuclide releases in the event of canister failure. Long-term containment within the canisters in turn depends primarily on the ***proven technical quality of the engineered barrier system (EBS)*** and ***favourable near-field conditions for the canisters.*** The technical quality of the EBS is favoured by the use of components with ***well-characterised material properties*** and by the development of appropriate acceptance specifications and design criteria. ***Favourable and predictable bedrock and groundwater conditions*** are requirements for selecting a waste disposal site.

The characterisation of the Olkiluoto site and the strategy for repository design are focused on a volume of bedrock situated between 400 and 700 metres below the ground surface, where favourable and predictable bedrock and groundwater conditions are expected to be found. Therefore, the safety concept automatically includes the depth requirement from YVL 8.4, instructing Posiva to locate the repository “*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 8.4, section 3.3).

Besides providing a protective environment for the canisters, the siting and design of the disposal system ensure that the transport of radionuclides released from an initially defective or subsequently breached canister will be effectively retained and retarded by the other barriers. These are provided by the secondary set of safety pillars in Figure 1-4, which comprise ***slow release from the spent fuel matrix, slow diffusive transport in the buffer, and slow radionuclide transport in the geosphere.***

The safety concept relies on a ***robust system design***. A robust design aims at a system whose performance is relatively insensitive to possible imperfections in its implementation and to the unavoidable residual uncertainties in the understanding its future evolution.

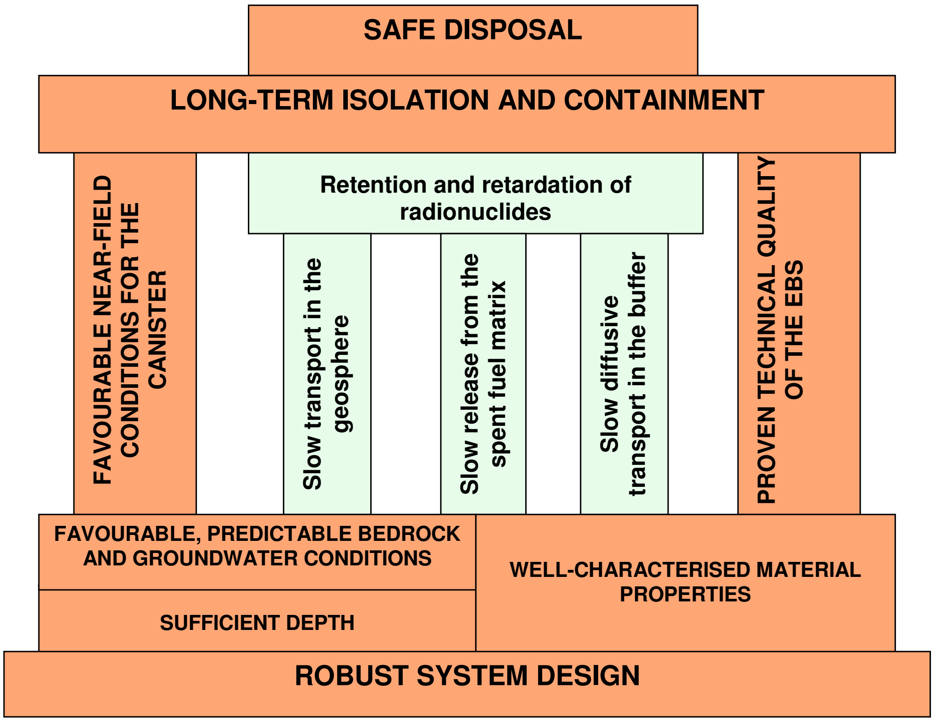

Figure 1-4. *Outline of safety concept for a KBS-3 type repository for spent fuel in a crystalline bedrock (adapted from Vieno & Ikonen 2005). The safety concept is based on a robust system design. Red pillars and blocks link safety features of the disposal system on which safety primarily depends to the overall goal of the safety concept (safe disposal). Green pillars and blocks indicate secondary safety features that may become important in the event of radionuclide release from a canister.*

1.4 Safety strategy

At present, Posiva's activities are aimed at achieving a sufficient maturity to submit an application for a repository construction license at the end of 2012. In essence, this means focusing on the following activities:

- design of the repository and encapsulation facility,
- development of the technology needed for implementing the KBS-3 design in such a way that it meets the safety requirements set by the Finnish authorities,
- research aimed at developing an understanding of the Olkiluoto site and assessing the long-term safety of a KBS-3 repository at the site, and
- demonstration of the safe operation of the disposal facility and development of the case for the long-term safety of disposal.

The ongoing construction of the underground rock characterisation facility, ONKALO, provides an important input to these activities.

In practice, carrying out the design, development and research work as parallel activities implies the need for iteration. The management of this iterative work constitutes Posiva's core process until the construction phase. An integral part of this process (see Figure 1-5) is the management of requirements at several hierarchical levels: at the top level there are the requirements for implementation coming from the producers of spent fuel on one hand, and the safety criteria coming from the authorities and guiding the implementation on the other. Below these, there are the technical requirements for the disposal system and its components.

As explained in TKS-2006 (Posiva 2006), the major goal of the present programme phase 2006-2009 is to define the performance targets (see Section 2.1.2) for the KBS-3 system components and demonstrate that these can be met by the technology available and by the properties of site selected for the repository. This goal leads to Research, Development and Technical Design (RTD) work following five main directions:

- design and technical demonstration of the disposal technology,
- development and application of the rock suitability criteria,
- development of site understanding,
- design of a repository layout adapted to the site features, and
- development of the safety case for the application of the construction license.

The aim is to achieve a high level of confidence in the long-term safety of the disposal system and demonstrated compliance with regulatory safety criteria and constraints.

The communication of this confidence and compliance is a major challenge in safety case reporting. A strong safety culture is essential to develop and maintain confidence in safety within the implementing organisation, and also to communicate this confidence to the regulator and to the general public. Safety culture means putting safety first in the pursuit of the strategy shown in Figure 1-5, especially at key decision points, such as those indicated by diamonds in the figure. To enhance confidence in long-term safety, such safety culture should be pervasive at all hierarchical levels within the implementing organisation and also acknowledged by outsiders.

1.5 Delivery of Safety Case

The safety case report portfolio from the Safety Case Plan 2005 (presented in Figure 1-3) was designed to convey the elements of the safety case, as defined in Section 1.1.

The main products of the present Safety Case Plan 2008 will also be organised in a report portfolio; however, to enhance the quality management of the safety case activities (see Chapter 5), some important changes will be introduced. The main purpose of these changes is to increase the traceability and transparency of the data production and modelling processes and to identify the critical data, parameters and assumptions that are used in the safety case.

The new safety case portfolio is shown in Figure 1-6. The Safety Case Plan 2008 on the top refers to the present report. The *Description of the Disposal System Report*

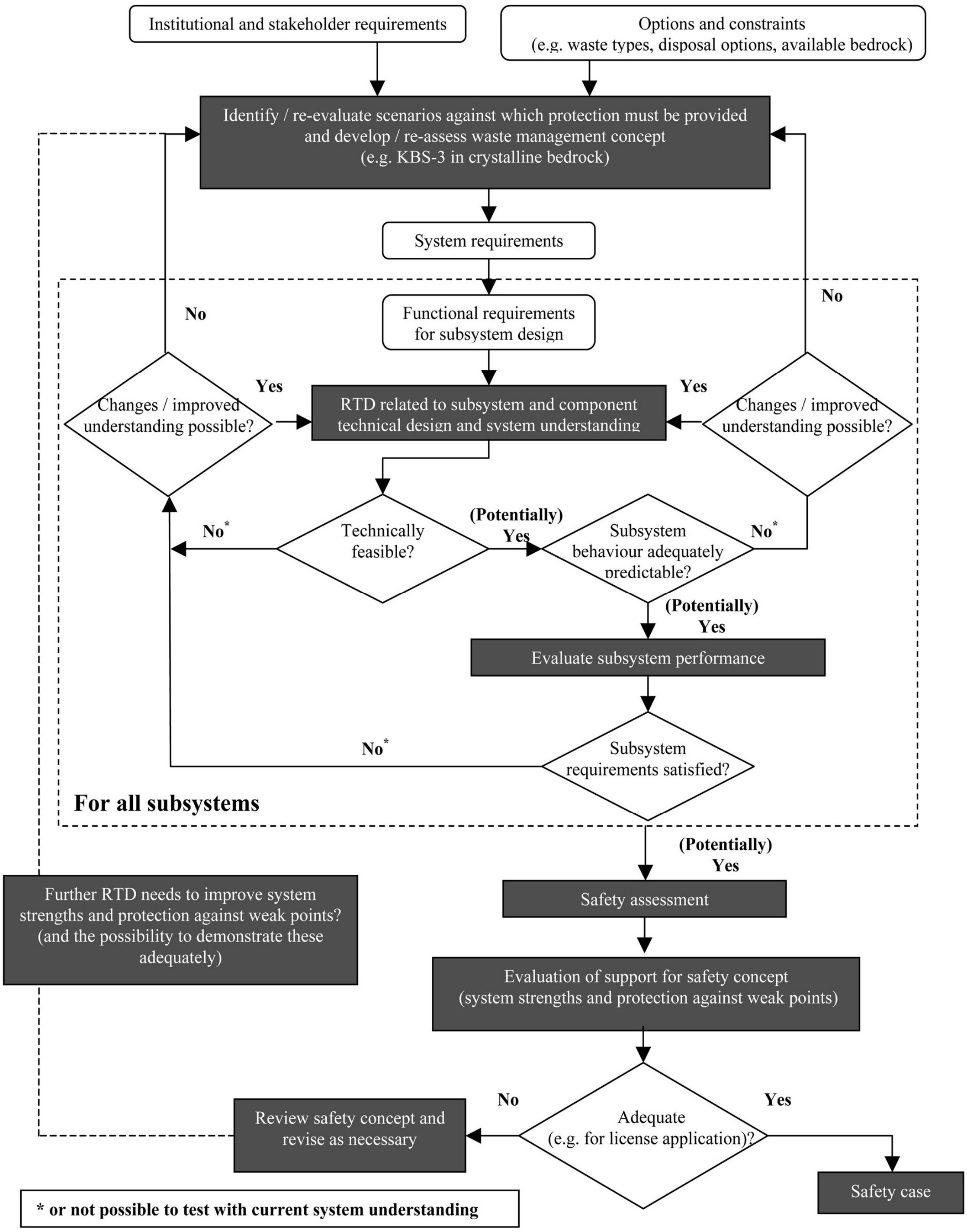

***Figure 1-5**. The iterative process followed in identifying Research, Development and Technical Design (RTD) needs and priorities according to Posiva's safety concept (Posiva 2003b, Figure 2.1).*

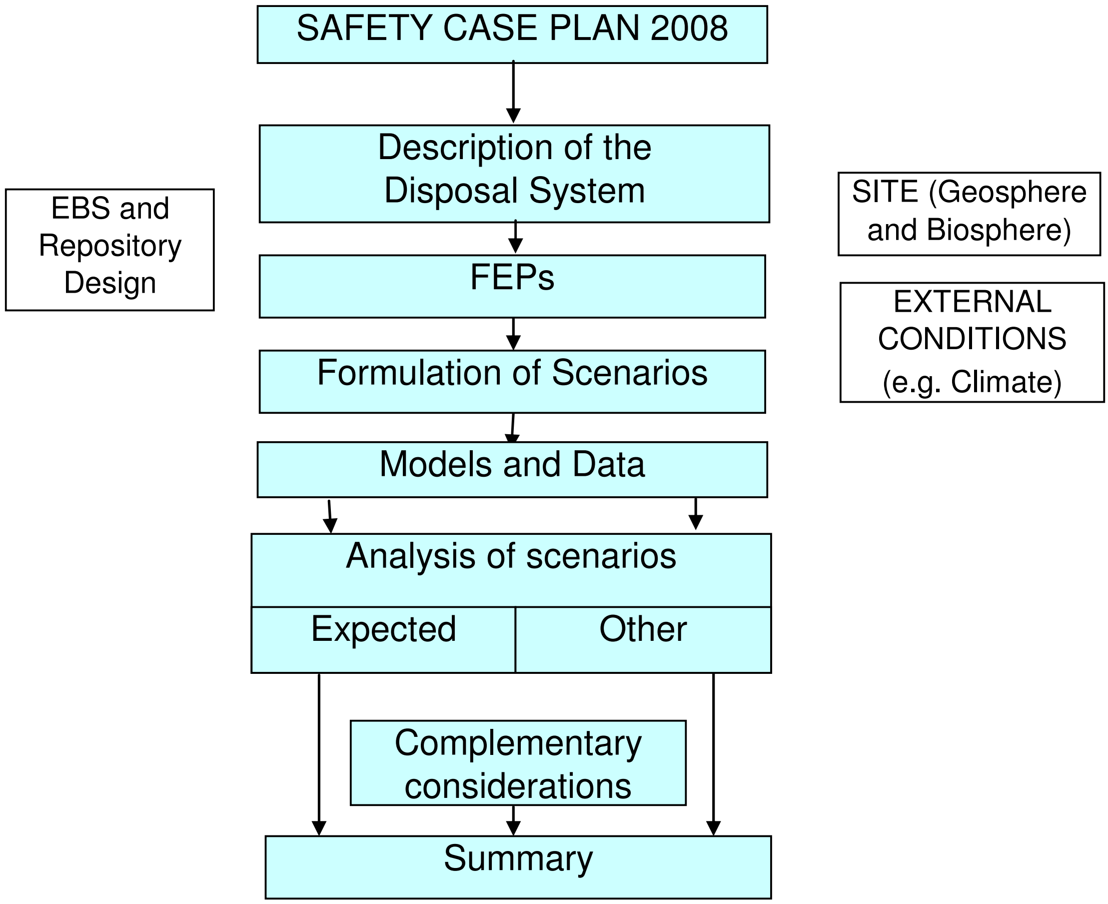

Figure 1-6. *Main reports of the new safety case portfolio (in blue) and the main input from supporting technical and scientific activities (in white).*

summarises the information on the waste form, the engineered barrier system and the Olkiluoto site. More detailed descriptions are given in technical and scientific reports on various components of the disposal system, including the site descriptive model of Olkiluoto and the description of biosphere conditions. Background analyses related to future climatic conditions will also be performed and reported.

In the new safety case portfolio, the earlier *Biosphere Assessment Portfolio* has been fully integrated. For example, the biosphere data and models are handled in the *Models and Data Report* and its supporting reports. However, the expression *biosphere assessment* will be retained. In the present report, it refers to the description of the current biosphere and its evolution, landscape modelling, and the assessment of radiological consequences.

The *features, events and processes* affecting the evolution of the repository will be described in the *Process Report* supported by a FEP lists or database background report. The evolution of the repository and the scenarios for analysis in the safety assessment will be described in the *Formulations of Scenarios Report*.

A key novelty in the new safety case portfolio is a *Models and Data Report* documenting the data and their interpretation (including modelling) in the context of the safety case. The *Models and Data Report* will be the main link between the safety case and the EBS design and development as well as between the safety case and the Olkiluoto site investigations and biosphere descriptions. The report will explain the most significant data used in the actual assessment. It will consist of a discussion of those parameters, data and underlying assumptions that are considered important for the results of the safety assessment calculations and the conclusions of the safety case as a whole. The scope and contents of the first *Models and Data Report* will be based on earlier safety assessments, but the report will later be revised to reflect new knowledge and lessons learned during the safety case process.

Scenarios to be analysed quantitatively will address both the expected evolution of the repository and disruptive events, as required by YVL 8.4. The assessment of the releases of radionuclides and the radiological consequences of these releases will be presented in the *Analysis of Scenarios Report*.

The *Complementary Considerations Report* (earlier Complementary Evaluations) is carried over from the earlier Safety Case Plan 2005. An example of such report was produced as a part of the KBS-3H safety assessment (Neall *et al*. 2007).

Finally, the whole safety case, including the main results, will be described in a *Summary Report*. This report will provide the main input to the Preliminary Safety Analysis Report (PSAR) needed for the application for a repository construction license.

The production of the safety case portfolio and the main activities in the safety case process as a whole will proceed iteratively, as shown in Figure 1-7. The production process is divided into four main sub-processes. The *conceptualisation & methodology* sub-process frames the assessment providing the description of the disposal system, FEPs and the formulation of scenarios, including the Olkiluoto- and design-specific description of the evolution of the disposal system. The *critical data handling and modelling* sub-process makes the central linkage between Posiva's main technical and scientific activities and the production of the safety case. The *assessment* sub-process analyses the consequences of the evolution of the disposal system in various scenarios, classified either as part of the expected evolution or as disruptive scenarios and analyses their potential consequences. The *compliance & confidence* sub-process is responsible for the final evaluation of compliance of the assessment results with the regulatory criteria and for overall confidence in the safety case. The safety case production process outlined in Figure 1-7 is the basis for the quality management of safety case activities as described in Chapter 5.

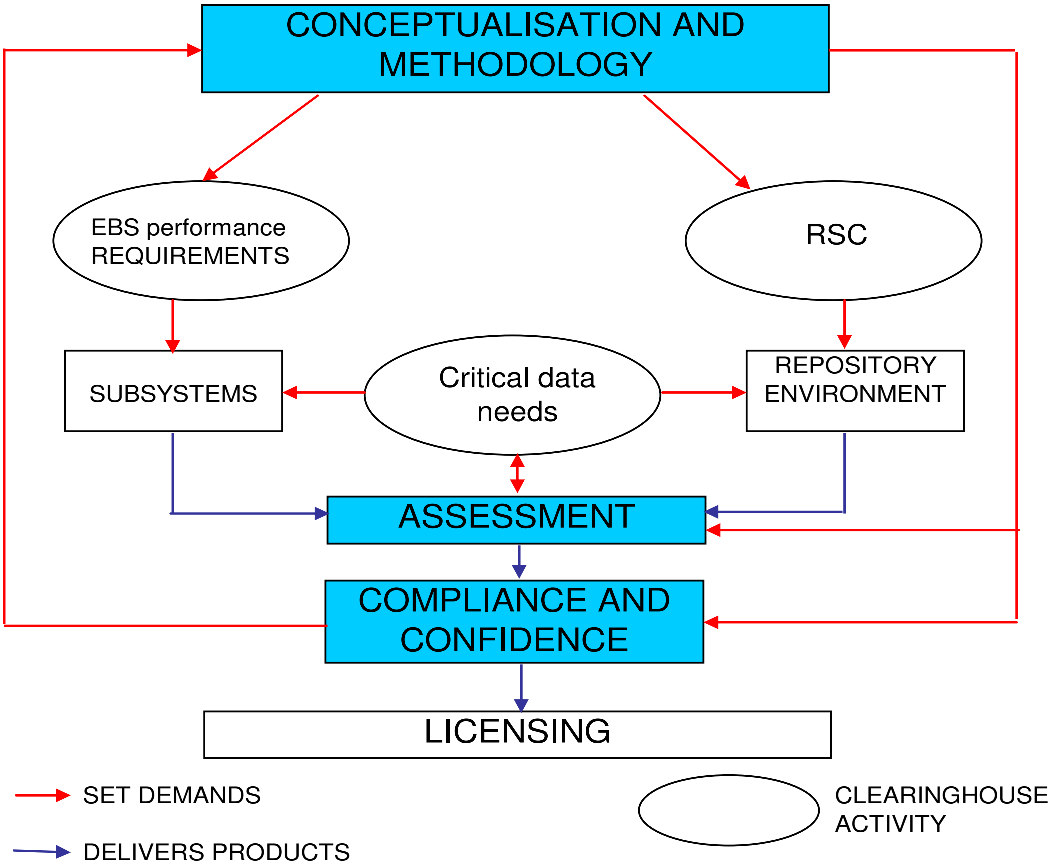

Figure 1-7. *Main activities of the safety case (production) process.*

The main elements of the new safety case portfolio and the methodology used in the safety assessment are discussed in Chapter 2. The supporting activities are described in Chapter 3. The management of uncertainties is outlined in Chapter 4. The principles of quality management for the safety case activities are provided in Chapter 5. Finally, the implementation of the new plan is briefly presented in Chapter 6.

2 MAIN ELEMENTS OF THE SAFETY CASE

2.1 Description of the disposal system

2.1.1 General description

A description of the disposal system is needed for the purposes of the safety assessment and the safety case. Currently, two variants of the KBS-3 concept are under consideration: KBS-3V & KBS-3H, as shown in Section 1.1, Figure 1-2. These variants, however, have similar components in the near field and an identical geosphere and biosphere at Olkiluoto, and so much of the description applies to both, although important differences are indicated where necessary.

The disposal system consists of the spent nuclear fuel, the canisters, the buffer, the tunnel backfill and tunnel plugs, the geosphere and the biosphere in the vicinity of the repository. The spent fuel canisters are disposed of either vertically in deposition holes (KBS-3V) or horizontally in larger-diameter 100-300 m long deposition drifts (KBS-3H) in a one-storey underground facility with disposal tunnels at a depth of about 420 m below ground.

In the KBS-3V and the KBS-3H variants, the basic components are largely similar in design. There are, however, important differences between the two designs are linked to the backfilling of the deposition tunnels (KBS-3V) and the plugging system for the deposition drifts (KBS-3H). The presence of additional steel components in KBS-3H drift is also an important difference compared to KBS-3V. In KBS-3H, each canister is pre-packaged in a special assembly, called a “supercontainer”, which consists of a perforated steel shell cylinder containing the canister, surrounded by bentonite clay buffer material. The purpose of the supercontainer is to facilitate emplacement operations. Bentonite distance blocks separate adjacent supercontainers one from another along each drift. Filling blocks are emplaced in drift sections in which high groundwater inflow makes them unsuitable for supercontainer and distance block emplacement. Highly transmissive fractures intersecting the drifts may be isolated using a system of steel compartment plugs.

The safety assessment and the safety case require a detailed description of the state of the repository at the time it is built (the initial state) and of the changes the repository will undergo at later times. In the case of KBS-3V, the initial state is defined as the conditions of one canister and deposition hole at the time of deposition (in the case of KBS-3H, it will include the state of the deposition drift in the vicinity of the canister). The discussion of initial state for all deposition holes (or all positions along the deposition drifts) is, however, fairly complex, since the initial state will be reached at different times in different parts of the repository; in other words, the canisters and deposition holes will be in different stages of their thermal, hydraulic, mechanical and chemical (THMC) evolution at any given time.

To emphasise the important role of the THMC processes taking place during the operational phase, when parts of the repository cavities are under construction or in operation, the term ***post-emplacement*** (instead of post-closure) assessment is used, indicating that the safety assessment and the safety case begins at the time the first canister is emplaced.

The initial state of the engineered parts of the repository system is obtained largely from the design specifications of the repository, including allowed tolerances for deviations. Furthermore, the manufacturing, construction and installation processes and the quality control measures applied to these activities have to be described in order to discuss and specify the various conceivable ways that the initial state may deviate from the design specifications. Demonstration activities to show the feasibility of construction and emplacement of the different components within the tolerances given are planned. A plan of demonstration activities in ONKALO will be published in late 2008.

The repository will be operated over a period of about 100 years. During this operational period, spent fuel canisters will be emplaced in deposition holes (or deposition drifts for KBS-3H), which will be backfilled and/or plugged as soon as possible to minimise hydraulic disturbances. For the same reason, the repository will be excavated in stages, thereby minimising the open volumes present at any given time.

The initial state of the geosphere and biosphere, as well as ongoing evolutionary processes, such as land uplift, groundwater flow and mixing of groundwater types, are determined by site investigations and compiled in the site descriptive models of the geosphere and the biosphere. These have been presented in the Olkiluoto Site Description 2006 (Andersson *et al*. 2007) and in the Olkiluoto Biosphere Description 2006 (Haapanen *et al.* 2007), and will be updated as site characterisation proceeds.

Current understanding of the characteristics of spent fuel has been summarised in Anttila (2005) and the current canister design is described in Raiko (2005). Repository design and layout reports have recently been compiled for both KBS-3V (Saanio *et al.* 2006, Kirkkomäki 2007) and KBS-3H (Autio *et al.* 2008, Johansson et al. 2007). See further Section 3.4. In addition to the canister and repository designs, the reports present the reasoning behind the design, as well as the basic dimensioning analyses and demonstrations of technologies.

It should be noted that the design of many parts of the disposal system (e.g. tunnel plugs) are still under development and future design changes are possible. The stepwise development of the repository and its design means that the degree of knowledge and detail of the design specifications will increase at each step, whereas uncertainties are expected to decrease.

All the components of the repository must meet the special requirements that are defined in Posiva´s requirements management system (VAHA). The *VAHA programme and database* (see Section 3.5) has been established to define, document and manage requirements placed on each component of the disposal system stemming from different origins (e.g. stakeholders, regulatory, engineering). The VAHA system thus includes the design basis of the disposal system, but it can also be used to assess the range of possible deviations that need to be addressed in the safety case.

2.1.2 Safety functions and performance targets

The safety concept for a KBS-3 repository is described in Section 1.3 and depicted in Figure 1-4. The concept is based on the long-term isolation and containment of the spent fuel assemblies in the canisters, but, in the event of radionuclide release from the canister, the rest of the engineered barrier system and the host rock shall also retain and retard the transport of the released radionuclides. In accordance with this safety concept, *safety functions* have been assigned to the EBS and host rock.

The main safety function of the canister is to ensure a prolonged period of complete containment of radionuclides. This safety function rests first and foremost on the mechanical strength of the canister insert and the corrosion resistance of the copper surrounding it.

The safety functions of the buffer include (a), protection of the canisters from external processes that could compromise the safety function of complete containment of the waste and associated radionuclides and (b), limitation and retardation of 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 preventing the possibility of preferential pathways for flow and advective transport along the buffer/rock interface. This is required because the buffer/rock interface may locally be perturbed by the steel supercontainer shell and its corrosion products.

The safety functions of the host rock are (a), to isolate the repository from the biosphere and normal human habitat, (b), to provide favourable and predictable mechanical, geochemical and hydrogeological conditions for the engineered barriers, protecting them from potentially detrimental processes taking place above and near the ground surface such that they contain the spent fuel, and (c), to limit and retard inflow to and release of harmful substances from the repository.

Other system components, including e.g. backfill, plugs, structural and sealing components have not been assigned safety functions. They are, however, designed to be compatible with, and support the safety functions of, the canister, the buffer and the host rock. For example, backfilling and sealing of the repository cavities support the safety functions of the host rock by preventing the formation of water conductive flow paths and by discouraging inadvertent human intrusion into the repository.

In its most recent safety assessment SR-Can (SKB 2006a), SKB also used the concept of safety functions and defined safety function indicators and safety function indicator criteria in order to identify and assess the critical safety components of the barriers. According to SKB's terminology:

- A safety function describes how a repository component contributes to safety. (Example: The canister should withstand isostatic load.)

- A safety function indicator is a measurable or calculable property of a repository component that indicates the extent to which a safety function is fulfilled. (Example: Isostatic stress in canister.)

- A safety function indicator criterion is a quantitative limit such that if the safety function indicator to which it relates fulfils the criterion, the corresponding safety function is maintained. (Example: Isostatic stress < isostatic collapse load.)

SKB's approach has recently been applied in the KBS-3H safety assessment using Olkiluoto as a reference site (Smith *et al*. 2007c).

Finnish regulations use the terms "performance targets" rather than safety function indicators, but, as discussed below, the meaning is similar. According to YVL Guide 8.4 (Section 3.2), the performance targets shall be defined for the long-term performance of each barrier:

"Targets for the long-term performance of each barrier shall be determined based on best available experimental knowledge and expert judgement. […]The determination of the performance targets for the barriers shall be based on an assumption that, due to some unpredicted phenomenon, the performance of a single barrier as a whole may be significantly lower than the respective target value. The safety requirements shall be met even in such case. The determination of the performance of barriers shall take account of changes and events that may occur in various assessment periods. […]The performance targets for the engineered barriers shall be set so that there will be no releases of radioactive substances into the host rock during the assessment period of several thousands of years".

The safety functions described above provide the starting point for developing the performance targets and design bases for the repository and EBS components. Performance targets and criteria for the system components will be specified in the *VAHA programme and database* mentioned above, which is discussed in more detail in Section 3.5. The safety functions for the barriers are derived from the disposal concept, whereas performance targets take into account design-specific issues. For example, material selection (e.g. buffer material) can set requirements on the rock environment, and vice-versa. Given a reference design and the knowledge of relevant material properties, a range of conditions under which the barriers function as they should can be defined and formulated as performance targets, similar to the safety function indicator criteria in SKB's terminology. To locate rock volumes where it is likely that the performance targets set for the EBS will be upheld, the *Rock Suitability Criteria (RSC-) programme* (see Section 3.5) has been set up. Such criteria are site specific, e.g. they consider the most relevant parameters and the confidence in the models at the given site, whereas the safety functions are in essence independent of the site.

It should be noted that the use of safety function indicator criteria, as defined in SR-Can, was found to have some limitations when applied to the KBS-3H safety assessment (these limitations may also apply in the case of KBS-3V). For example, SR-Can defined no indicators or criteria for the buffer/rock interface, although uncertainties in the properties of this interface were considered at length in the KBS-3H safety assessment. Furthermore, the criteria do not distinguish between local changes in, for example, buffer density, and overall changes in average buffer density. To overcome such limitations, it is likely that performance targets and related criteria will be refined iteratively over time based on experience from their application in safety assessment and design studies.

2.2 Features, Events and Processes

The ways in which of features, events and processes (FEPs) are treated either explicitly or implicitly in the safety case must be adequately documented. The main report of the safety case portfolio (Figure 1-6) dealing with FEPs is the *Process Report*. The *Process Report* describes the features of the repository and the site and the events and processes related, for example, to the geographical position of the site (i.e. climate, and climate changes), along with their relevance in the various time frames of interest for the safety case. A new background report on the current FEPs database will be produced in 2010 and will include a list of all the FEPs relevant to a KBS-3 repository located at the Olkiluoto site, including those for the biosphere.

Posiva's first Process Report on the KBS-3V concept was issued in 2004 (Rasilainen 2004). In this report, processes are classified according to whether they are radiation-related, thermal, hydraulic, mechanical, chemical, or are relevant to radionuclide transport. A new classification scheme was introduced in the updated Process Report by Miller and Marcos (2007) (see Figure 2-1). Process classification in this report is based on the relevance of each process to either system evolution or radionuclide transport. The new classification scheme allows the interaction between processes to be taken into account while avoiding unnecessary repetition in the description of the processes. The significance of each process is described according to the time frame within which it operates and its relevance to radionuclide containment, transport and retention. Furthermore, the FEPs discussed are cross-checked for completeness against

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

A Process Report for a KBS-3H repository has been jointly issued by Posiva and SKB (Gribi *et al.* 2007). The KBS-3H Process Report provided the basis for the Evolution Report (Smith *et al.* 2007a). The processes discussed were selected on the basis of the comparison with the KBS-3V processes discussed in SR-Can, according to a "difference analysis" approach. Thus, the focus was mostly 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. Since most of these differences concern the early evolution of the buffer and the additional steel components in the drift (e.g.

HANDLING OF PROCESSES

Category: indicates the relevant sub-system component (e.g. canister) and type of process (evolution or migration related)

General description: provides concise explanation of the process and current understanding of how it operates

Olkiluoto-specific issues: provides a summary of any site specific issues that need to be considered

Uncertainties: overview of main conceptual and parameter/data uncertainty

Time frames of interest: relevance to post-operational, re-equilibrium or glacial period

Scenarios of relevance: relevance of process to climatic scenarios, base scenario or assessment scenarios

Significance: for the long-term containment of radionuclides (RN) within the canister or for RN transport or retention after canister failure

Treatment in performance assessment: how the process is dealt with in RN transport and performance assessment models (implicitly, explicitly)

NEA's equivalent FEP: correspondence of the process to process/es in the NEA international FEP database (NEA 2000)

Figure 2-1. *Template for the handling of processes described in Miller & Marcos (2007), process significance and relationship to scenarios.*

supercontainer steel shell, steel plugs), near-field processes and evolution were discussed in more detail than those of the far field (see Gribi *et al*. 2007 and Smith *et al*. 2007a).

Biosphere-specific processes are not described in these earlier reports. There are, however, a number of processes relevant only to the biosphere that are taken into account at least implicitly in the Biosphere Description by Haapanen *et al*. (2007), in terrain and ecosystem development modelling (Ikonen *et al*. 2007), and in landscape modelling (e.g. Broed *et al*. 2007). These processes will be included in the next update of the *Process Report*, and described in detail in various background reports.

The approach to be used for the classification and treatment of processes in the next update of the *Process Report* will take account of the experience gained in compiling the previous Process Reports and related documentation for the KBS-3V and KBS-3H concepts. As mentioned above, a FEP list or database background report will be produced in 2010 for traceability purposes.

2.3 Formulation of scenarios

2.3.1 Background and overview

A safety case must consider the range of possible future conditions and/or events that could influence the evolution of the repository over time. The *Formulation of Scenarios Report* will describe possible paths for the evolution of the repository over time, and identify scenarios to be evaluated quantitatively in the analysis of scenarios. 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*. 2007a).

In accordance with Section 28 of the Government Decision, "*compliance with the long-term radiation protection objectives as well as the suitability of the disposal concept and site shall be justified by means of a safety analysis that addresses both the expected evolutions and unlikely disruptive events impairing long-term safety*" (STUK 1999). In YVL 8.4, STUK provided guidance on the contents of such a safety analysis, which includes "*the analysis of the potential future evolutions of the disposal system (scenarios analysis)*" (STUK 2001).

According to the International Atomic Energy Agency (IAEA), a scenario is "*a postulated or assumed set of conditions and/or events. They are most commonly used in analyses or assessments to represent possible future conditions and/or events to be modelled, such as possible accidents at a nuclear facility, or the possible future evolution of a repository and its surroundings*" (IAEA 2003).

YVL Guide 8.4 (STUK 2001, Section 4.2) states that "*A scenario analysis shall cover both the expected evolutions of the disposal system and unlikely disruptive events affecting long-term safety. The scenarios shall be composed systematically from features, events and processes, which are potentially significant to long-term safety and may arise from*

- *mechanical, thermal, hydrological and chemical processes and interactions occurring inside the disposal system*

- *external events and processes, such as climate changes, geological processes and human actions.*"

Posiva's methodology for scenario formulation is to follow a top-down approach, i.e., start from the regulatory safety requirements concerning scenario analysis in YVL 8.4 and then consider how FEPs and associated uncertainties affect the various safety functions of the disposal system and their evolution over time.

The evolution of the disposal system will be affected by the FEPs external to the system, and, in particular, evolving climatic conditions at the site. Climatic conditions affect the likelihood of events including major earthquakes that could affect the engineered structures of the repository (such earthquakes are unlikely before the end of the next ice age).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. Climatic scenarios are therefore formulated to provide a framework within which the internal evolution of the disposal system can be described. The climatic scenarios used in the most recent safety assessments are the Weichselian-R and the Emissions-M scenarios, as defined in the Evolution report (Pastina & Hellä 2006). Posiva has recently started a new project on the formulation of alternative climate scenarios for use in future safety assessments.

Having defined the climatic evolution scenarios, the scenarios for the evolution of the disposal system itself are formulated. These describe how mechanical, thermal, hydrological and chemical process and interactions identified in the *Process Reports* (see Section 2.2) affect the evolution of the disposal system and its safety functions over time.

Following regulatory guidance a "base scenario" as well as "variant" and "disturbance scenarios" are defined in order to assess the consequences of the expected evolution and those in the case of unlikely disruptive events. The relationship between these classes of scenarios and the framework set by the climatic scenarios is illustrated in Figure 2-2 (note that the assessment scenarios correspond to the variant and disturbance scenarios in YVL Guide 8.4).

The definition of the base scenario is described in Section 2.3.2. The variant and disturbance scenarios are discussed in Section 2.3.3. System evolution in each scenario must also be described in detail as a basis for the analysis of consequences, and measures taken to ensure that no important FEPs have been overlooked. These aspects are discussed under "analysis of scenarios", in Section 2.5.

2.3.2 The base scenario

According to Guide YVL 8.4, *"The base scenario shall assume the performance targets defined for each barrier, taking account of the incidental deviations from the target values"*.

Performance targets for the system components (e.g. canister, buffer, host rock) will be specified in the VAHA programme and database and criteria will be used to define the ranges of conditions under which the performance targets will be achieved (see Section 2.1.2). The repository is sited and designed to ensure that the criteria are fulfilled in its initial state. Furthermore, if the system evolves as expected, the criteria should continue to be fulfilled over a prolonged period, irrespective of minor deviations from target values of parameters such as buffer density and swelling pressure. Posiva's interpretation of the base scenario, as defined by STUK, thus corresponds to the expected evolution of the repository during which the canisters will isolate the waste and provide complete containment of radionuclides over at least a million year time frame. No radionuclide releases are expected to occur in this scenario during the time period over which quantitative assessment is considered meaningful.

Even though it involves no releases, in the base scenario the disposal system will undergo significant changes over time, particularly in the early, transient phase when the heat output from the fuel is high and the buffer and near-field rock are undergoing

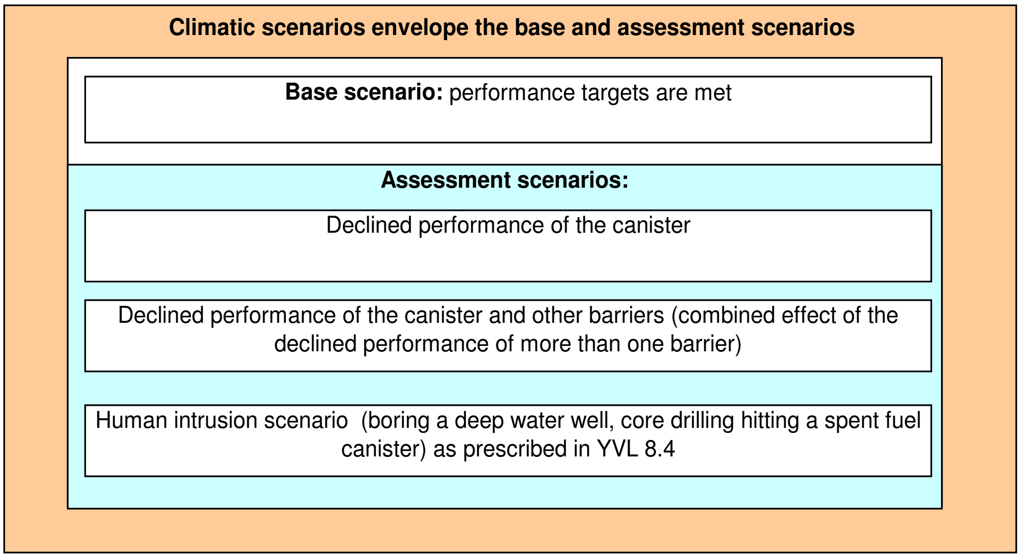

Figure 2-2. *Scenario classification in the safety case.*

saturation. The description of the base scenario will include an account of the system´s ***post-emplacement evolution*** that starts at the time of emplacement of the first canister in the repository and extends to the far future, explaining why performance targets and related criteria are expected to remain fulfilled.

2.3.3 Variant and disturbance scenarios

In addition to the base scenario, YVL 8.4 directs Posiva to assess the consequences of variant and disturbance scenarios: *"The influence of the declined overall performance of a single barrier or, in case of coupling between barriers, the combined effect of the declined performance of more than one barrier, shall be analysed by means of variant scenarios. Disturbance scenarios shall be defined for the analysis of unlikely disruptive events affecting long-term safety.*"

Disruptive events to be considered that are specified in YVL Guide 8.4 are:

- *"boring a deep water well at the disposal site;*
- *core drilling hitting a spent fuel canister; and*
- *a substantial rock movement occurring in the environs of the repository"*.

Posiva's methodology to formulate variant scenarios addressing the declined overall performance of one or more barriers, as well as any additional disruptive scenarios, is as follows:

1. Consider the safety functions of each of the main components of the disposal system.
2. For each safety function, identify and derive one or more performance targets and related criteria (e.g. buffer density range).

3. Develop understanding of the system and its evolution - with a focus on the safety functions.
4. Identify the failure modes (loss of safety functions) that could occur in the course of system evolution.
5. Consider if and when the occurrence of such failure modes is plausible.
6. Consider the implications of loss of one safety function on the others.
7. Identify plausible descriptions of the evolution of safety functions over time (the scenarios to be analysed in the safety assessment).
8. Use FEP lists, such as NEA's International FEP database (NEA 2000), to check for scenarios completeness.

The methodology is similar to that applied in the KBS-3H safety assessment, which is in turn based on that developed and applied in SR-Can (SKB 2006a). It uses the concepts of safety function indicators and safety function indicator criteria (analogous to performance targets and related criteria - see Section 2.1.2). In the KBS-3H safety assessment, the safety functions and their associated indicators and criteria were used to identify which of the uncertainties and detrimental phenomena identified in the Process and Evolution Reports had the potential to disturb the safety functions of the main system components significantly, and thus give rise to variant scenarios. The example of the safety function indicators and corresponding criteria for the canister and buffer used in the KBS-3H safety assessment and in SR-Can is shown in Figure 2-3. The three safety function indicators and corresponding criteria for the canister correspond to the three canister failure modes: corrosion, isostatic failure, and rupture due to rock shear. The evaluation of safety function indicators was made by means of scoping calculations and qualitative arguments.

The scenario formulation methodology will also include the use of scoping calculations to assess the impact of uncertainties and detrimental phenomena on the safety functions of the components and to determine whether or not the performance targets are met. Scoping calculations can be used, for example, to evaluate the effects of perturbations to the buffer / rock interface on the corrosion rate of the copper canister, and hence on the canister safety function indicator of "copper thickness". Scoping calculations can also address the effects of transient processes such as piping and erosion during saturation of the buffer that could potentially affect the performance targets and related criteria for the buffer safety functions.

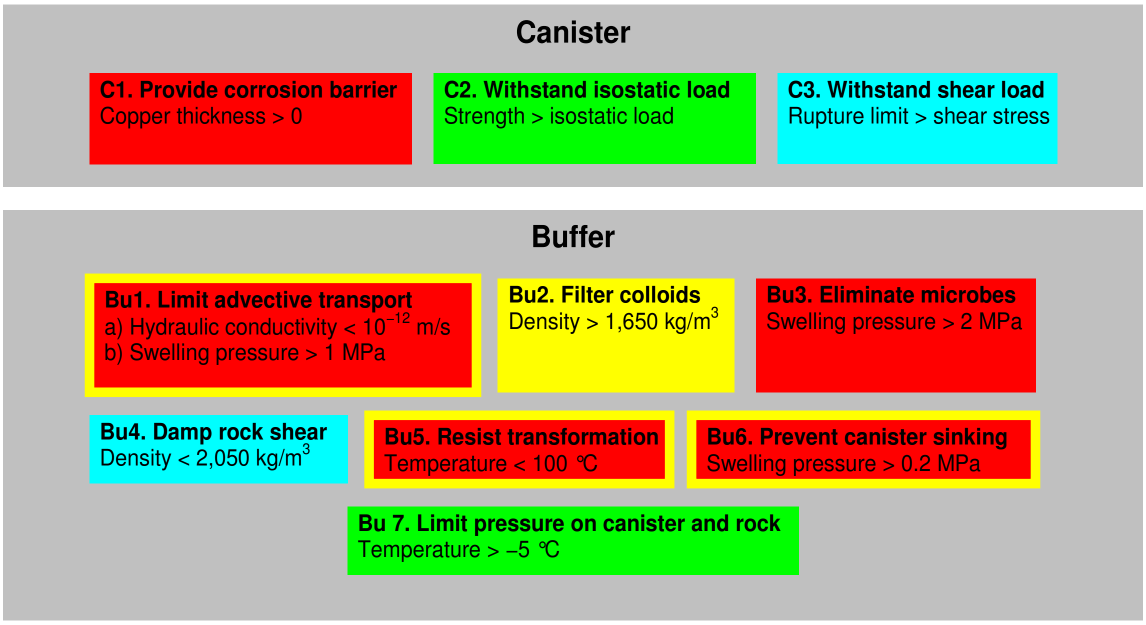

Figure 2-3*. Safety functions and safety function indicator criteria for the canister and buffer, after SR-Can (SKB 2006a).*

2.4 Models and data

2.4.1 Objectives

The *Models and Data Report* summarises the models and key input data used in the safety case for describing evolution, radionuclide transport and dose assessment. It includes models and data related to the spent nuclear fuel, the EBS and the site (including both geosphere and biosphere). It is the main outcome of the sub-process *critical data handling and modelling* as introduced in Section 1.5.2 and further described in Section 5.4.3.

The objectives of the *Models and Data Report* are manifold:

- provide an overview of the models used in support of the safety case, including the models used in the description of the site and repository evolution (both expected evolution and disruptive events), release and transport analysis and assessment of radiological consequences,
- collect and document the key input data used in the safety case and present them in a form that can be reviewed by independent peers,
- describe the data production process from “raw data” to the “analysis of scenarios input data”, to enhance the traceability and transparency in the chain of generating, selecting, justifying and documenting the data for the safety case,
- describe uncertainties, their impact and how they are treated in the safety case,
- describe the review and quality control procedure of key input data,
- document where expert judgement and expert elicitation processes have been applied (see Section 5.6.1) and present the basis for final data selection,

- identify data/model gaps and any requirements for data/model updating, and
- guide RTD and quality management activities to focus on the most critical parts of the data production process.

As the objectives of the *Models and Data Report* suggest, the main motivation for including this report in Posiva's safety case portfolio has been to facilitate the quality management of the safety case (see Sections 1.5.2 and 5.4.3) and the treatment of uncertainties (see Chapter 4). Specifically, the *Models and Data Report* will provide the main documentation of the models applied in developing the safety case and of the input data for these models. Guide YVL 8.4 states that such documentation shall be included in the documentation supporting the Preliminary Safety Analysis Report, which will be submitted with the construction license application.

A critical step in the implementation of the *Models and Data Report* is the identification of the specific models and data to be included. Inclusion of models and data in the report is based on their relevance to the evaluation of the post-emplacement performance of the EBS (including geosphere properties affecting the behaviour of the engineered barriers), and to the evaluation of radionuclide transport and dose assessment in the event of canister failure. Spent fuel data and data on engineered and residual materials (e.g. steel and cementitious materials) that are not considered part of the EBS and data concerning the interfaces between barriers will also be included, whenever these may affect barrier performance. The information given will not be restricted to design or baseline data. Data for the different states of the system will also be presented, including the initial state and taking into account the effects of the construction and implementation and the reference conditions during the repository and site evolution. Possible deviations from the expected initial conditions due to implementation mishaps will also be discussed based on the information presented in design reports and technology demonstration activities and on judgements regarding the effectiveness of the quality control measures applicable in the implementation phase. Models and data for both variants of the KBS-3 concept, KBS-3V and KBS-3H, will be presented.

It is planned to publish the first *Models and Data Report* at the end of 2009 and an updated version will be published in 2012. In the 2009 version, the models and data used thus far in the most recent safety assessments (Nykyri *et al*. 2008, Smith *et al*. 2007c) will be documented. Other data used in the broader safety case will be included if considered relevant to the assumptions made regarding repository and site evolution. The 2009 version will also aim to identify any data needed for the PSAR that is still lacking. It will thus be used for the planning of activities in the period 2010-2012, which will be presented in the TKS-programme 2009. The updated (2012) version of the *Models and Data Report* will document the input database for the safety assessment to be carried out in support of the PSAR. The actual parameter values used in the different calculation cases will be reported as part of analysis of scenarios. The parameter values will be selected from the ranges of possible values or probability distributions given in *Models and Data Report*.

2.4.2 Main contents of the Models and Data Report

According to current thinking, the *Models and Data Report* will consist of three parts:

Part I-General: this part will cover issues common to both the EBS and the site (geosphere and biosphere), such as the general background and introductory information. It will include a short description of the Olkiluoto site and the repository concept, with repository layout, main dimensions and spent fuel inventory data. The planned repository construction, operation and closure schedules will also be presented here.

Part II-Models: this part will present an overview of the conceptual models and numerical codes used in the analysis of scenarios taking into account system evolution, radionuclide release and transport and radiological consequences. The overview will point out the links between the different models and the data flow between them, e.g. how the results of hydrogeological and hydrogeochemical modelling are transferred to the assessment of the near-field performance or how the geosphere and biosphere radionuclide transport models are related.

Part III-Data: this part will contain the discussion on data related to the components of the EBS, including data on spent fuel, engineering and residual materials, the interfaces between barriers, the geosphere and the biosphere required as input for the models described in Part II.

The *Models and Data Report* will not present new information concerning the EBS and the site. The data are based on the information presented in supporting documents published by Posiva or elsewhere in the literature. These may include reports on site properties and EBS design, the evolution of the site and repository, manufacturing, operation and quality control procedures, and the results of the demonstration activities. The activities and projects providing key input data to be used in the safety case and for design are briefly described in Chapter 3.

2.4.3 Models

Models may take the form of more qualitative, conceptual models, or numerical models, the governing equations of which are generally solved using computer codes. The discussion of the models in the *Models and Data Report* will cover:

- conceptual models, how the models are related to the FEPs and the evolution of the repository system in different scenarios,
- the role of the models in analysing or producing information to be used either directly or indirectly in the evolution description or transport modelling, and in showing that the safety functions of the system and the performance targets of the barriers are fulfilled,
- conceptual uncertainties related to model abstraction – e.g. geometrical simplifications, effective parameters used by models and model limitations – including relevant time frames and conditions of application,

- information about the codes used, such as the names of the codes, the general principles and conditions of application and key input parameters, and
- verification and validation aspects, such as whether the code is a commercial product or has been developed for a specific application and how it has been verified and validated.

Three main types of models (both conceptual and numerical) are used in the safety case: 1) models used for process and evolution understanding, 2) models used in radionuclide release and transport calculations and 3) models used in assessing radiological consequences. These different model types are discussed individually below in the context of the *Models and Data Report.*

Models used for process and evolution understanding

Several models are used to describe processes and to analyse the evolution of the disposal system, as described, for example, in the KBS-3V Process Report (Miller & Marcos 2007). Examples of such models include the source term, the thermal, rock mechanics, groundwater flow models and the several models describing the thermal, hydraulic, mechanical and chemical processes affecting the disposal system. An important part role of these models is to analyse the interfaces between the different barriers, how the barriers interact with each other and the impact of these interactions on system evolution. These analyses should also consider issues related to the implementation and operation of the repository, e.g. operational schedule, the impact of design features, gaps between different components and the evolution and impact of any residual materials.

Biosphere processes (e.g. ecological processes) are also evaluated or more qualitatively assessed using several models, such as the land uplift model, the surface and near-surface hydrological model, erosion and sedimentation models, vegetation models, fauna habitat and future human activity models.

Models used in radionuclide release and transport calculations

Near-field release and transport modelling, geosphere transport modelling and biosphere transport modelling are carried out separately in the analysis of calculation cases in safety assessment (see Section 2.5). Near-field release and transport modelling addresses the release of radionuclides from spent fuel and transport within the canister, the buffer and the tunnel backfill and between these. Geosphere transport modelling addresses transport of released radionuclides through fractures is in the host rock. The release from the near field to flowing groundwater in rock fractures intersecting the deposition hole (KBS-3V) or deposition drift (KBS-3H) is modelled by applying a transfer coefficient at the interface, as described in the recent assessments (Nykyri *et al*. 2008, Smith *et al*. 2007c). Biosphere modelling converts the radionuclides releases from the geosphere to appropriate endpoints.

Each model considers a range of processes. As an example, the modelling of near-field release and transport generally includes:

- modelling of releases from the several spent fuel types, each with different release functions;

- advective and / or diffusive transport within a system of engineered barriers;
- sorption on solid surfaces;
- solubility limits; and
- radionuclide decay and ingrowth.

Geosphere transport modelling considers groundwater flow and transport of substances along fractures. In the rock matrix domain (the rock adjacent to the fracture), the phenomena modelled include diffusion and sorption on solid surfaces. Radioactive decay and ingrowth are represented in both domains, and transfer of radionuclides across the boundary between the domains takes place by diffusion. A revised concept for radionuclide transport modelling in the geosphere has been presented by Poteri (2007) and will be applied in the upcoming safety assessments. The concept is based on discrete fracture network modelling and results in flow paths with variable transport resistance and retention properties along them. The analysis considers a large number of single flow paths and dispersion is not explicitly considered in the models. This is considered to be a conservative approach. The impact of dispersion along individual paths may be studied separately, if considered necessary, as was done in TILA-99 (Vieno & Nordman 1999, p. 165).

Modelling of radionuclide transport in the biosphere (landscape modelling) consists of a time-dependent linked transport model containing the biosphere objects (forests, wetlands, lakes, rivers, coastal and agricultural areas that might possibly receive even indirectly, radionculide releases from the repository). These objects are represented by ecosystem-specific compartment models. The connections between the objects are derived from terrain forecasts for the period from the present to the end of the assumed time period when regulatory dose constraints apply.

The models described briefly above are likely to be applicable in the majority of calculation cases evaluated in the analysis of scenarios in the safety assessment. Other models may, however, also be required to handle special cases or scenarios, such as transport of radionuclides in volatile form by repository-generated gas and human intrusion scenarios.

Models used in assessing radiological consequences

Radiological consequences to humans are assessed using the dose concept described in Avila & Bergström (2006). In 2005, a revision of the methodology for dose calculations was carried out within a joint project commissioned by SKB and Posiva. This included a compilation of dose coefficients for ingestion and inhalation and an update of the dose coefficients for external exposure. Physiological characteristics to be used in dose calculations, such as water intake, food intake and inhalation rate, were reviewed and summarised. The model does not use the traditional approach of making assumptions concerning living habits and exploitation of the landscape. Instead, a reference man approach is used, according to which humans need a given food energy intake, expressed as total carbon intake, derived from reference values of protein, carbohydrate, and fat intake (ICRP 1975, 2004).

The assessment of radiological consequences to other biota will mainly be based on the integrated approach proposed in the Environmental Risk from Ionising Contaminants:

Assessment and Management (ERICA) project, which was conducted under the EC (EC (European Commission´s) 6th Framework Programme (Beresford *et al.* 2007). ERICA proposed an integrated approach to scientific, managerial and societal issues concerned with the environmental effects of contaminants emitting ionising radiation, with emphasis on biota and ecosystems. The assessment will follow a stepwise approach with three-level criteria to differentiate cases in which the effects on biota are low or negligible from those in which there is a need to assess the consequence using all relevant information available.

2.4.4 Data

Following the approach used in SR-Can's data report (SKB 2006b), the discussion of specific data will cover:

- Source of information (e.g. from laboratory or in situ tests, and related modelling). Specific emphasis will be put on the conditions for which data are supplied with respect to their intended use.
- Impact of information on assessment results, including any sensitivity analyses or bounding calculations that may have been performed to evaluate how the system evolution and ultimately the radiological consequences are affected by the data at hand.
- Data uncertainty, including spatial variability (e.g. scaling issues, discrete features) and temporal variability (e.g. variation of groundwater composition over time).
- Correlations among parameters used in the safety assessment (e.g. influence of water-rock interaction on groundwater composition).
- Quantification of the parameter values, considering the most likely values and ranges or distributions of the values. According to the discussion in Chapter 4, probability distributions for certain parameters will be given when there is a sufficient basis to support them; otherwise best estimate values and ranges of variation covering the most "pessimistic" values consistent with current scientific understanding will be given. The quantification process takes into account uncertainty related to data (assessed by the technical experts) and scenarios considered in the safety assessment (assessed by the safety assessment team), see Figure 2-4.
- Description of the quality assurance and quality control activities.

Data will be organised according to the system component to which they relate: spent fuel, cast iron insert and copper canister, buffer, backfill, geosphere and biosphere. Implementation related data covering engineering and residual materials and including relevant information related to excavation and operation will also be included. These various groups of data are discussed individually, below.

Spent fuel data, insert and copper canister

Examples of data related to spent fuel, the copper canister and the canister insert to be included in the *Models and Data Report* include:

- Copper and cast iron physical and mechanical data.

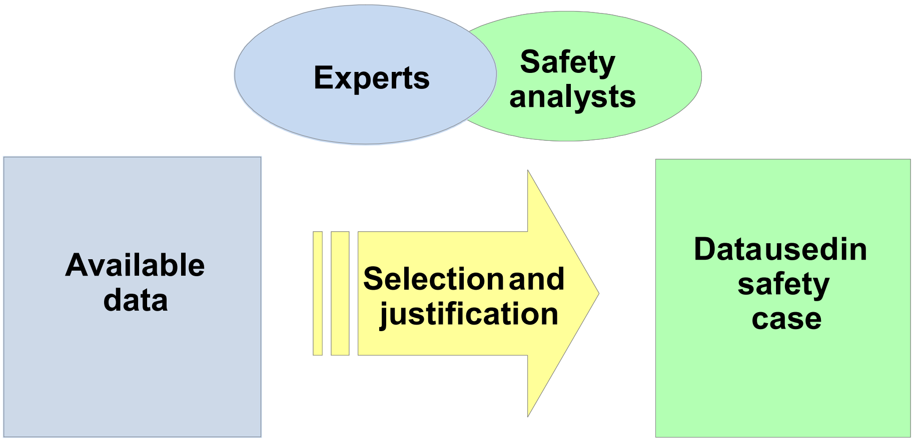

Figure 2-4. *The Models and Data Report describes the process of selection and justification of data used in the safety case; this process is based on the interaction between the technical experts and the safety analysts.*

- Copper and cast iron corrosion parameters under different physico-chemical conditions expected during canister evolution.
- The dimensions and evolution with time of possible or hypothetical canister defects, which are needed for radionuclide release and transport calculations.
- Radionuclide inventories.
- Radionuclide partitioning between fuel matrix, instant release fraction, zircaloy and other metallic parts.
- The fractional dissolution rate of the fuel matrix and the corrosion rates of the zircaloy cladding and other metal parts.
- Criticality data.
- Solubilities of radionuclides relevant to the safety case.

Some background information related to inventory data and canister and insert data is given below.

Inventory to be considered in the safety case

The existing repository plans have been made for the disposal of three different fuel types (BWR, VVER 440 and EPR) from five nuclear power reactors: Loviisa 1 and 2, Olkiluoto 1, 2 and Olkiluoto 3 using. Based on the current planned operational lifetimes of Loviisa 1 and 2 and of Olkiluoto 1, 2 and 3, and with the current maximum discharge burnup, this corresponds to approximately 5 500 tU in 2 840 canisters (Chapter 3 in Miller & Marcos 2007). However, owing to TVO's and Fortum Power and Heat's plans to build more nuclear capacity both at Olkiluoto and Loviisa, Posiva has started an Environmental Impact Assessment (EIA) Process to raise the maximum capacity of the repository to 12 000 tU. In the future the discharge burnups of the spent fuel elements may also become higher which would decrease the amount of the spent fuel but increase its thermal output per kgU. The assumptions concerning the radionuclide inventories and other spent fuel specifications will be decided on the basis of the status of plans in 2010-2011.

Canister and insert data

Much of the current information about the canister and the insert is presented in the Canister Design Report (Raiko 2005), an update of which will be published by the end of 2008. Key issues for the safety assessment are the probability, characteristics and evolution of canister defects. Posiva is currently developing manufacturing and inspection technologies to estimate the probability of initial canister defects. Reports describing inspection and manufacturing technologies, including seal weld technology, will be published by the end of 2009.

A related issue is the data required for modelling radionuclide release from a canister with a penetrating defect. Release involves several interacting processes (including the evolution of the defect over time, water intrusion, corrosion of the cast iron insert, gas generation and release, and the thermo-hydro-mechanical-chemical evolution of the system of the defective copper-iron canister and buffer).

Buffer

Examples of buffer data to be discussed in the *Models and Data Report* include:

- Physical and chemical properties.
- Thermal properties.
- Hydraulic properties.
- Mechanical and rheological properties.
- Transport data.
- Geometrical dimensions.

MX-80 sodium bentonite from Wyoming, USA has been the reference buffer material in Posiva's previous safety assessments. 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 from Milos, Greece, is also currently under consideration for reference.

Currently, many buffer design details remain open. A Buffer Design Report will, however, be published by the end of 2009.

Backfill

A description of the backfill including physico-chemical information, volumes and masses as well as their location in the repository will be included in the *Models and Data Report*.

The types of data presented for the backfill will be similar to those presented for the buffer. However, different types of backfill (e.g. crushed rock/bentonite mixtures, swelling clays) and different possible physical forms (e.g. pellets, blocks) will be taken into account. Posiva is currently developing a site-specific backfilling concept within the Coordination Programme for the Olkiluoto-Specific Backfilling (OBA), as described in Section 3.3.

Implementation-related data

Implementation-related data are data not only about materials but also about repository construction and operation. Implementation-related materials are referred to as engineering and residual materials. Engineering materials for which data will be compiled include:

- Structural materials, associated, for example, with iron-bearing structures outside the canister, such as the supercontainer shell and compartment plugs used in KBS-3H.
- Sealing materials, such as cementitious materials or colloidal silica (Silica-Sol) used for groundwater control and for construction purposes.
- Plugs, such as structures made of cement, rock, bentonite (or a combination of the above) used to isolate the deposition tunnels, the central tunnel, ramps and other entrances from the surface and to plug boreholes.

Residual materials from construction and disposal operations (e.g. organic additives in cement grouts, organic materials from vehicles and human waste and oxidising by-products from drilling and blasting) may also be left in the repository after closure.

Engineering and residual materials are important to long-term safety because of their potential to influence the near-field chemical environment, and hence affect important barrier functions and/or near-field conditions, such as radionuclide solubility and sorption behaviour, concentrations of corrosive species, and the abundance of nutrients for microbial activity. For example, the iron-bearing structures outside the canister in KBS-3H can affect the bentonite safety functions due to the possible alteration of montmorillonite and formation of new minerals (Wersin *et al.* 2007). Corrosion of iron-bearing structures also generates hydrogen gas, which may affect the saturation time of the buffer, as discussed in the KBS-3H Process Report (Gribi *et al.* 2007). Cement-bearing structures, such as plugs, or groundwater control techniques using cement, may also affect the buffer safety functions if highly alkaline cement leachates react with the montmorillonite.

Data concerning repository construction and operation include design dimensions and tolerances, dimension and nature of the gaps between the different barriers and nature of the interfaces between different barriers. Potential implementation mishaps involving more than one barrier (e.g. buffer damage during installation) will also be discussed in this section, while potential implementation mishaps specific to one barrier will be discussed in the barrier-specific sections of the report (e.g. canister defects during welding).

Because the plans for closing and sealing the repository are still under development, the first *Models and Data Report* will focus mostly on information used in the most recent KBS-3V and KBS-3H safety assessments. It is expected that the 2012 update will contain more detailed information on the engineering and residual materials, as the design develops further.

Geosphere

The current conditions of the Olkiluoto site, described in terms of geological, rock mechanics, hydrological, hydrogeochemical and transport models, are given in the

Olkiluoto Site Descriptions (e.g. Posiva 2005, Andersson *et al.* 2007), which are regularly updated. The evolution of the site conditions has been discussed by Pastina & Hellä (2006) and updates will be given in the planned *Formulation of Scenarios Report* (Section 2.3). The *Models and Data Report* will focus on data applied in the models for describing the main processes relevant to the safety case, especially those affecting canister integrity, buffer behaviour and radionuclide transport. The data discussed will therefore include those related to:

- Groundwater composition, including the presence of gases and microbes.
- Thermal properties.
- Fracture data, including impact of fracture minerals on the buffering capacity and retention properties of the rock.
- Rock mechanics.
- Groundwater flow.

The currently ongoing excavation of ONKALO, the construction of the repository and its subsequent operation will disturb the natural properties of the rock. Examples of these disturbances include e.g. the introduction of engineering and residual materials which will affect groundwater composition, and the formation of the excavation damaged zone (EDZ), which will affect the hydraulic properties of the rock / buffer interface. The impact of construction and repository operation on the properties of the geosphere will be considered in defining the parameter values to be used in the safety case.

Biosphere

The biosphere assessment aims at describing the past, present and future conditions of the surface systems of the Olkiluoto site, track the fate of any radionuclides released from the repository to the biosphere, and assess possible radiological consequences on humans and other biota. From a modelling and data point of view, the biosphere assessment can be divided into three main components: *deriving terrain forecasts*, *radionuclide transport modelling in the landscape*, and *assessing radiological consequence*. The *Biosphere Description* is an important background process feeding these components with information. The most important data sets (and related models) of each component are:

Terrain forecasts:
- Digital elevation model (DEM).
- Land uplift data and model.
- Climate scenario, especially regarding sea level and hydrological balance.
- Sedimentation and erosion model.

Landscape modelling:
- Geometrical properties of biosphere objects.
- Sorption properties of soils and sediments.
- Ecological properties - mainly root uptake and subsequent translocation.
- Hydrological properties of each biosphere object.

Consequence analysis:

- Anatomical and physiological properties of humans and dose-response data for both humans and other biota. Note that these are, however, taken as given in the biosphere assessment, since international standards and recommendations e.g. from ICRP and the FREDERICA database are utilised.
- Agricultural statistics and consumption data.
- Food and water intake by humans and representative fauna.
- Geometrical properties and habitation data of representative fauna.

2.5 Analysis of scenarios

Section 2.3 describes how scenarios will be formulated, including a base scenario and a number of variant and disturbance scenarios. The base scenario represents the most likely evolution of the repository, with no release of radionuclides from the canisters over a million year time frame. The variant and disturbance scenarios, which are collectively termed assessment scenarios, include those postulating a defective canister and those dealing with other features of the system, disruptive events and processes that might also lead to canister failure and radionuclide release. The following sections describe how the consequences of radionuclide releases arising in the assessment scenarios will be analysed, taking into account all relevant uncertainties. The uncertainties include those related to the performance of the barriers. This is in agreement with Section 3.2 in Guide YVL 8.4 (STUK 2001), which states that performance "*may diverge from the respective target value due to rare incidental deviations such as manufacturing or installation failures of engineered barriers, random variations in the characteristics of the natural barriers or erroneous determination of the characteristics*". The comparison of the calculated radiological consequences of the assessment scenarios with regulatory requirements is also discussed. This will take account of the estimated probability of occurrence of the scenarios and will differ between "expected scenarios" that are judged to have a significant probability of occurrence and "other scenarios" that arise from unlikely disruptive events.

2.5.1 Repository calculation cases

Parameter and model uncertainties result in a range of possible evolutionary alternative paths for the repository and its environment in each scenario. These alternative paths are analysed in sets of *repository calculation cases* (called scenarios in the terminology of TILA-99) using mathematical models. Parameter and model uncertainties in the biosphere assessment give rise to biosphere-specific scenarios, treated separately in *biosphere calculation cases*.

Model parameters quantify the characteristics of the phenomena that influence repository evolution, including the rates of processes, the timing or frequency of events and the spatial extent of features. These are typically subject to parameter uncertainty, such that there are ranges of values that are possible according to current scientific understanding. Different combinations of values from the possible values presented in the *Models and Data Report* (see Section 2.4) will be selected in a systematic way.

Uncertainties may also give rise to alternative model assumptions within a given scenario, and the impact of these will also be explored by defining multiple repository calculation cases. An example from the KBS-3H safety assessment is the uncertain transport properties of the buffer/rock interface. Considering the expected evolution, perturbations to the interface caused, for example, by the presence of a steel supercontainer and its corrosion products were assumed to have a negligible impact on mass transfer across the interface. In an alternative and pessimistic scenario, however, the interface was treated as a highly conductive “mixing tank”. Calculations were performed to explore the impact of these alternative interface models, both on canister lifetime and on radionuclide release to the host rock in the event of canister failure.

2.5.2 Deterministic and probabilistic approaches

In previous Finnish safety assessments, such as in TILA-99 (Vieno & Nordman 1999), a purely “deterministic” approach to the analysis of scenarios has been adopted, whereby the models and data that define each repository calculation case were individually specified. In future safety assessments, the emphasis is likely to remain on deterministic calculations. 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, as was done in safety assessments carried out by other repository programmes, for example, in the Swedish SR-Can (SKB 2006a) and in the Swiss Project Opalinus Clay (Nagra 2002) assessments. The aim will be to combine the merits of both approaches. For example, probabilistic calculations provide a systematic approach to exploring all possible parameter combinations. 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).

2.5.3 Deriving and structuring repository calculation cases

A systematic, transparent and consistent approach is needed for deriving repository calculation cases for each assessment scenario. Different sequences of events or processes may have similar consequences for the safety functions of the repository, and thus may be analysed in the same calculation case. It is therefore important to identify and organise identified formulation scenarios (Section 2.3) and calculation cases in a rational manner that avoids unnecessary computation.

In the recent Radionuclide Release and Transport Report for a KBS-3V type repository (Nykyri *et al.* 2008), the organisational structure shown in Table 2-1 was adopted[1]. According to this structure, the influence of the declined performance of the canister

[1] In the KBS-3H safety assessment, a somewhat different organisational structure was adopted, but the calculation cases themselves largely overlap. The exception is the human intrusion scenarios shown in Table 2-1, which were excluded from the KBS-3H safety assessment on the grounds than their impact would not be significantly different for a KBS-3H repository compared with a KBS-3V repository.

(Defective Canister Scenario DCS) is analysed in two variant scenarios, DCS-I and DCS-II. In STUK's nomenclature, DCS is an expected scenario. The combined effects of the declined performance of more than one barrier are analysed in the additional scenarios (AD). These scenarios comprise three variants that account also for disturbances to the geosphere and the buffer, and for the effect of repository generatd gases (see Table 2-1). Note that a declined performance of the buffer, as considered in AD-II, could be the result of different sequences of processes and events, including both initial misemplacement of the buffer and much later intrusion of glacial meltwater. In STUK's terminology, AD scenarios correspond to disturbance scenarios (other scenarios in Figure 1-6). Finally, two human intrusion scenarios, HI-I and, HI-II are considered, as prescribed by YVL 8.4.

Table 2-1. *Assessment scenarios for a KBS-3V type repository (modified from Miller & Marcos 2007).*

Assessment Scenarios	Descriptions and divisions into variants	Evolution of safety functions over time
Defective canister scenario (**DCS**)	**DCS-I**: delayed penetrating defect – radionuclide release starting at 10 000 years after the repository closure	**Canister:** failure from t = 10 000 years **Buffer, backfill, rock:** safety functions provided throughout assessment period
	DCS-II: early penetrating defect – groundwater in contact with spent fuel at the repository closure	**Canister:** failure from t = 0 **Buffer, backfill, rock:** safety functions provided throughout assessment period
Additional scenarios (**AD**) based on deviations in initial conditions and timing of internal or external processes	**AD-I**: Earthquake / Rock shear: Canister fails as a consequence of the sudden displacement of a fracture intersecting the deposition hole.	**Canister:** failure at different postulated times **Buffer, backfill, rock:** safety functions significantly perturbed due to rock shear
	AD-II: Canister fails as a consequence of disruptive events affecting the buffer: e.g. initial misemplacement of the buffer, intrusion of diluted meltwater	**Canister:** failure at different postulated times **Buffer:** safety functions significantly perturbed from t = 0 or from t = 100 000 years **Backfill, rock:** safety functions provided throughout assessment period
	AD-III: Gas expels water with IRF and/or gas (C-14) from the canister and deposition hole. The buffer and backfill have no role in the retention of radionuclides	**Canister:** failure from t = 0 **Buffer, backfill, rock:** safety functions for this specific radionuclide significantly perturbed from t = 0
Human intrusion scenario (**HI**)[2]	**HI-I**: Boring of a deep water well at the disposal site	**Canister, buffer, backfill, and rock:** failure or significant perturbation at different postulated times
	HI-II: Core-drilling penetrating into a canister	**Canister, buffer, backfill, and rock:** failure or significant perturbation at different postulated times

[2]The human intrusion scenario is to be dealt with in the Biosphere Assessment part of the Safety Case

For each scenario variant, individual repository calculation cases are defined to explore sensitivity to different parameter values and model assumptions and illustrate the impact of the various uncertainties in knowledge and data (Figure 2-5).

In Nykyri *et al*. (2008), a tree-structure methodology was used to formulate repository calculation cases for each scenario variant. The methodology facilitates the traceability of assumptions and data used in each repository calculation case. One assessment scenario is identified as being the most likely scenario for radionuclide release and, for this scenario, one case is identified as being the most realistic. This provides a base case against which to compare the results of other cases.

Figure 2-6 shows the tree-structure organisation of repository calculation cases for the specific scenario variant of a spent fuel canister with a penetrating defect (DCS-II). Defect time (t) refers to the time at which water is assumed to enter the canister through the defect leading to fuel matrix dissolution (assumed t = 0 for all cases in the recent KBS-3V safety assessment). Cases are grouped, at the highest level, according to the postulated defect size. A defect that escapes quality control would have to be small, and on this basis it can be stated that cases DCS-II.1 to DCS-II.4 are more likely than those assuming a larger defect. The larger defect cases are included to account for uncertainties in quality control procedures and, as such, are considered sensitivity analyses cases. Some of the cases identified by this methodology can, however, be excluded from quantitative consideration on qualitative grounds - i.e. on the basis of the knowledge of the evolution of the conditions at the repository site.

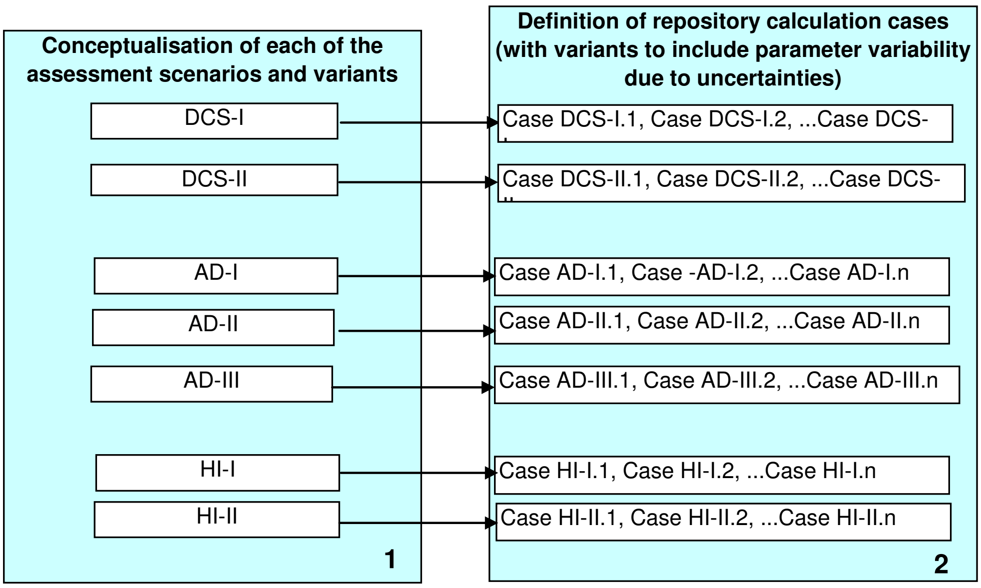

Figure 2-5. *Assessment scenarios and variants (1), for each of which repository calculation cases are defined that explore sensitivity to different parameter values and model assumptions and illustrate the impact of the various uncertainties in knowledge and data (2).*

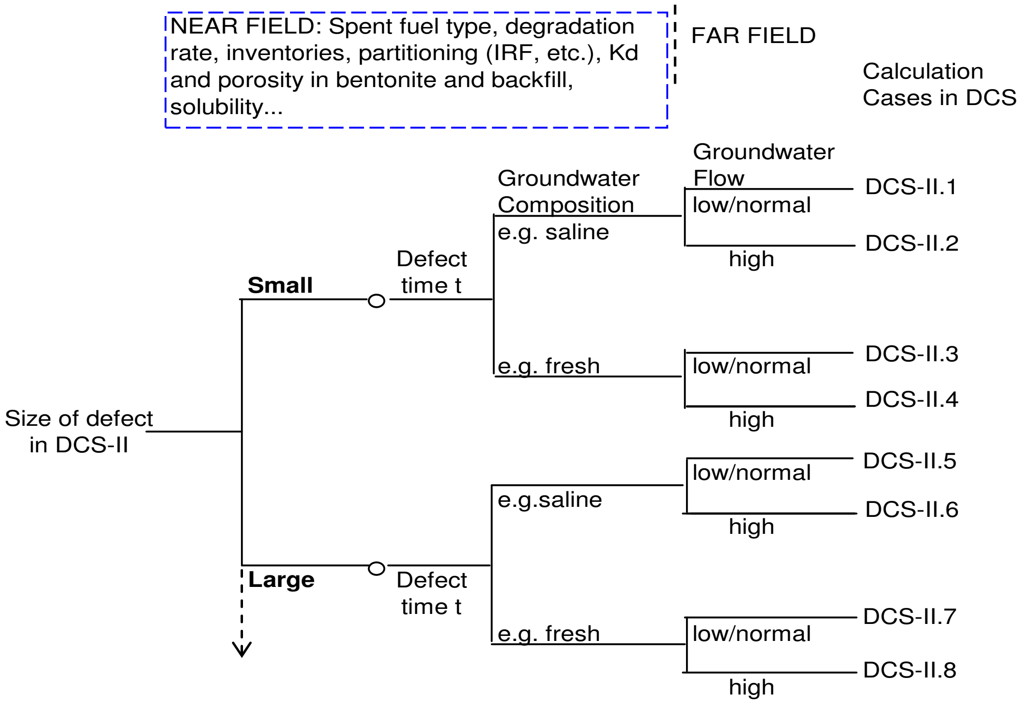

Figure 2-6. *The tree structure of the repository calculation cases in the Defective Canister Scenario used in the KBS-3V Radionuclide Release and Transport Report (Nykyri et al. 2008), where it is assumed that one or more canisters has either an initial penetrating defect of variable sizes (DCS-II).*

This same methodology for identifying the choices made in defining each calculation case can be extended to any of the scenarios to be analysed, allowing documentation of the rationale for the choices made in defining each case can be provided. A comparison of the methodology and results with the earlier TILA-99 Safety Assessment has been carried out in the recent KBS-3V Radionuclide Release and Transport Report (Nykyri *et al.* 2008) to explain differences in results and ensure consistency in system understanding.

2.5.4 Modelling release rates to biosphere and simplifying assumptions

The tree structure and associated descriptions shown in Figure 2-6 give mostly qualitative definitions of each case. For example, for DCS-II.1, a low/normal value will be used for groundwater flow, and the selected solubility limits and distribution coefficient (K_d) for buffer and backfill will correspond to saline water. Mathematical models, the governing equations of which are generally solved using computer codes, and input parameter values selected from the ranges and distributions given in the *Models and Data Report* provide a more precise definition of each repository calculation case for quantifying radionuclide release rates to the biosphere. Simplifying assumptions will, where appropriate, be used to avoid treating poorly understood phenomena or poorly defined uncertainties explicitly. Such assumptions - which can include the omission of some phenomena - are typically argued on the basis of supporting calculations or qualitative arguments either (a) to have negligible impact on performance, or (b) to be conservative, in that they lead to over-estimates of contaminant releases and their consequences.

In analysing each repository calculation case, measures will be taken to ensure the quality of application of models, datasets and computer codes. Quality control and assurance of the assessment, including validation of input data and assessment codes for their intended application, are addressed in Section 5.4. An important element of ensuring the quality of the assessment is the selection of the "numerical parameters" that are used, for example, to control the accuracy of an approximate numerical solution to a set of equations or the spatial or temporal limits of such a solution. Numerical parameters must also be carefully chosen as part of quality assurance.

2.5.5 Biosphere calculation cases

The results of the analyses of assessment scenarios are release rates at release locations (derived from groundwater flow modelling), which are used as input to the biosphere assessment. The release pattern is the distribution of release points at the surface that, depending on the release time, can correspond to a lake or forest or to any other object in the biosphere. The way in which various combinations (indicated with **X**) of release patterns and landscape model configurations give rise to different biosphere calculation cases is illustrated in Figure 2-7. The definition of each biosphere calculation case includes the specification of transport and exposure pathways. The process of building up transport and exposure pathways for the calculation of endpoints in the biosphere assessment is illustrated in Figure 2-8.

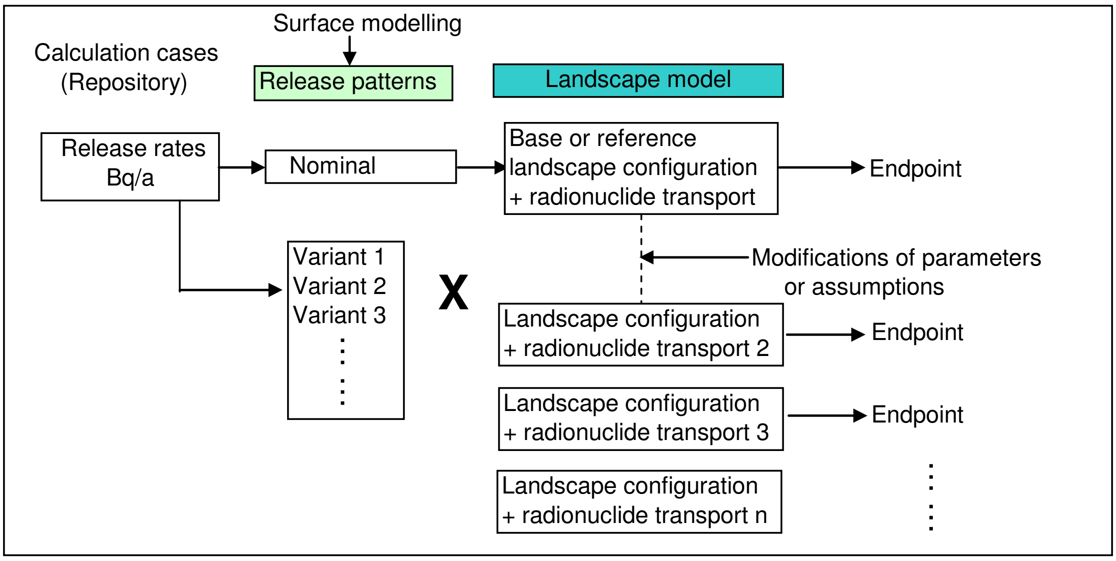

Figure 2-7*. The combination (X) of various possible release patterns and landscape configurations form alternative future paths in the radionuclide transport calculations in the biosphere assessment.*

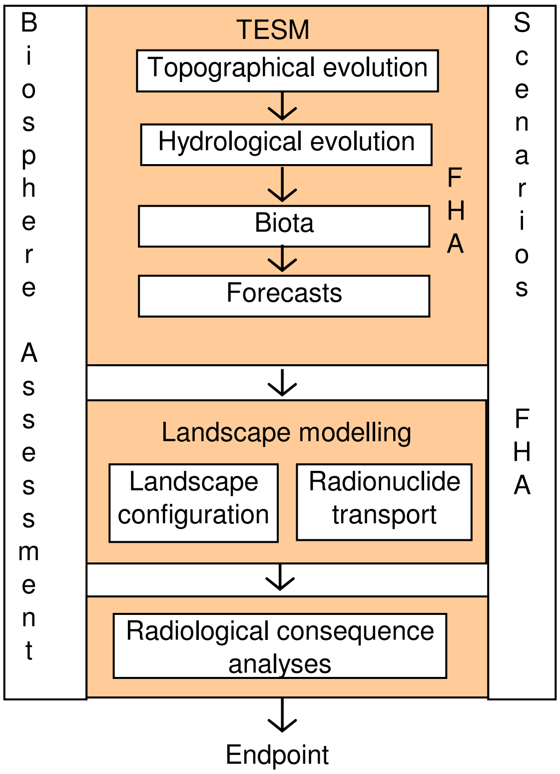

Figure 2-8. *Process of building up transport and exposure pathways for the calculation of endpoints in the biosphere assessment.*

The transport pathways are based on terrain and ecosystem development modelling (TESM) and landscape modelling (LSM). Future human activities (FHA) can be included in the build-up of transport pathways, if required.

The main results of biosphere calculation cases are time-dependent spatial distributions of the radionuclide activity concentration. These are used as input for the assessment of radiological exposure, which produces endpoints (or quantities) used to demonstrate compliance with regulatory constraints.

2.5.6 Comparison of results with regulatory requirements

Calculated radionuclide releases to the biosphere and the endpoints of biosphere assessment will be used to test compliance with regulatory requirements. As noted earlier (see Figure 1-6), scenarios will be divided into "expected scenarios" and "other scenarios", depending on their likelihood of occurrence. The expected scenarios include the base scenario, with no release of radionuclides from the canisters over a million year time frame, and any further scenarios that are judged to have a significant probability of occurrence. Other scenarios include those due to unlikely disruptive events. In the case of expected scenarios that lead to radionuclide releases and the associated calculation cases, the results of the analyses will be compared with Finnish regulatory dose or activity release constraints on the presumption that these scenarios will indeed occur (they are, in effect, assigned a probability of one). In the case of other scenarios due to unlikely disruptive events, the probability of occurrence will also be taken into account in assessing compliance. Guide YVL 8.4 (STUK 2001) states that: "t*he importance of safety of any such incidental event shall be assessed and whenever practicable, the resulting annual dose or activity shall be calculated and multiplied by the estimated probability of its occurrence*". In order to satisfy these regulatory requirements, the expectation value should remain below the radiation dose or activity release constraints in Guide YVL 8.4. If, however, the resulting individual dose implies deterministic radiation impacts (dose above 0.5 Sv), the order of magnitude estimate for its annual probability of occurrence *"shall be 10^{-6} at the most*."(YVL 8.4, Section 2.4).

2.6 Complementary considerations

Regulatory Guide YVL 8.4, Section 4.4, (STUK 2001) outlines the purpose and contents of the complementary considerations as follows:
"The importance of safety of such scenarios that cannot reasonably be assessed by means of quantitative analyses shall be examined by means of complementary considerations. The may include, e.g. bounding analyses by simplified methods, comparison with natural analogues or observations of the geological history of the disposal site. The significance of such considerations grows as the assessment period of interest increases, and the judgement of safety beyond million years can mainly be based on the complementary considerations.

Complementary considerations shall also be applied parallel to the actual safety analysis in order to enhance the confidence in results of the whole analysis or a part of it".

Complementary considerations can include, for example:

- observations and analyses from natural and anthropogenic analogues that build or support understanding of specific key processes (e.g. palaeohydrogeology and palaeoclimatology evidence and observations of the environment around ice sheets and permafrost areas) and of total system performance,
- comparison of the elements of the safety assessment methodology and results with earlier assessments, such as TILA-99 and SR-Can (in *the Analysis of Scenario Report)*, as well as other international safety assessments, whenever relevant, to ensure completeness, consistency and reasonableness of the present assessment,
- evaluation of complementary safety indicators that may be less affected by some uncertainties (e.g. uncertainties in future human lifestyles and also in geological processes on very long timescales) than effective dose and activity releases,
- consideration of the calculation results from a wider perspective and, in particular, the significance of their impact compared with other risks.

These topics usually lie outside the scope of quantitative safety assessments, but can nonetheless make important contributions to the safety case.

Based on such considerations, a Complementary Evaluations of Safety report (Neall *et al*. 2007) was produced as a part of the KBS-3H safety assessment. The report gathered together these diverse types of evidence and arguments related to long-term safety in order to promote confidence in the arguments, models and data used in the safety assessment. The report was structured according to the guidelines in YVL 8.4 and those in the Safety Case Plan 2005 (Vieno & Ikonen 2005). It also took into account the international consensus on the nature of these complementary considerations that are relevant to the post-closure safety case for geological repositories (NEA 2004a). The evaluations presented in Neall et al. (2007) will be supplemented in the coming years with results from activities at ONKALO, improved scientific understanding of processes and data acquired from natural analogue studies.

A *Complementary Considerations Report* for a KBS-3 type repository will be published in 2011. As mentioned above, some of the complementary considerations, such as the comparisons of elements of the safety case methodology in other national programmes as well as the main conclusions with the earlier safety assessment results will be presented in the *Analysis of Scenarios Report*.

2.7 Summary of the safety case

The *Summary Report* will summarise the contents of the whole safety case portfolio and provide the main input to the licensing process. Its final contents will depend on the pending revised regulations, but current structure is as follows:

1. Purpose and objectives of the safety case
    - as specified by the regulations
2. Main safety criteria
    - as also specified in the relevant regulations
3. Main components of the disposal system and their implementation
    - based on the *Disposal System Description Report*
4. Safety concept and safety functions
    - based on the material described in Chapter 1 and in Section 2.1
5. Assessment basis
    - based on *Models & Data*, *Process (FEPs)* and *Formulations of Scenarios Reports*
6. Summary of scenario assessment results
    - based on the *Analysis of Scenarios Report*
7. Summary of uncertainty analysis
    - based on the *Formulations of Scenarios, Models & Data*, and *Analysis of Scenarios Reports*
8. Summary of completeness considerations
    - based on *Process* (*FEPs), Formulations of Scenarios and Analysis of Scenarios Reports*
9. Statement of compliance
    - based on the *Analysis of Scenarios Report*

10. Evaluation of confidence
    - based on the above-mentioned portfolio reports and *Complementary Considerations Report*
11. Conclusions regarding necessary further work
    - based on the previous sections.

This structure is also broadly in agreement with safety case summary reports published by other repository programme implementers (e.g. Nagra 2002, SKB 2006a, Andra 2005, ONDRAF/NIRAS 2001) and follows the international guidelines and on the reporting of the safety case for geologic repositories (NEA 2004a).

A first version of the summary report for a KBS-3 type repository will be compiled in connection with the outline of the licensing documentation in 2009. An updated version will be published in 2012, in support of the construction license application.

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3 SUPPORTING ACTIVITIES

The supporting activities producing the main input for the safety case portfolio reports shown in Figure 1-6 are briefly described in this section. These activities focus on the description of the external conditions, site (including the biosphere) characterisation and understanding, EBS design and development as well as repository design, construction and operation. The additional supporting activity of requirements management, described in Section 3.5, manages the EBS performance requirements, which guide the design and development of the EBS and the plan for the overall repository implementation, as well as the rock suitability criteria (RSC), which establish the goals and directions for the site investigations and site modelling activities (see Figure 1-7).

3.1 Description of the external conditions

The location of the repository deep in the bedrock should decouple the disposal system from most processes occurring above it at the surface, but major changes in the surface conditions may also affect the conditions deeper down. The most important driver for such changes in external conditions is glacial cycling, which it is expected will continue to occur in the future. The formulation of possible future climatic states is, therefore, an important part of the safety assessment (the role of climatic evolution scenarios as a framework for discussing the evolution of the disposal system is described in Section 2.3).

Although it is not possible to predict exactly how the climate will change in the future, the range of more likely evolutions can be bounded using palaeoenvironmental information applicable to the Olkiluoto geographical local and regional position. Climatic scenarios are to be explored and selected climate sequences of particular interest for the safety case will be studied in detail to establish the context for the development of models used in safety analysis. The selected climate sequences in the Evolution Report (Pastina & Hellä 2006) were based on the work of Cedercreutz (2004) summarising the available literature on future climate scenarios and applying them to Olkiluoto. The climatic scenarios will be revised in the light of new work reported in the literature and comments received on the earlier work. For this purpose a major study has recently been initiated in cooperation with the Finnish Meteorological Institute.

There are several factors affecting climate change that have relevance in different time frames. In the time frame up to 1 000 years, the most important factors are atmospheric autovariation (natural and anthropogenic emissions of CO_2 and methane), atmosphere-ocean feedbacks, solar variability, air-sea-ice-land feedbacks and volcanic activity (dust clouds). In the time frame up to 10 000 years, ocean circulation and orbital parameters also play a significant role. After that, the most relevant additional factors are orogeny/isostacy, continental drift and polar wandering, evolution of the Sun, and variations in the concentration of galactic dust.

In all these time frames, the most important factors influencing climate change are:
- natural changes in greenhouse gases concentration,
- volcanic activity and natural aerosols, and
- anthropogenic activities (greenhouse gases and aerosols).

Climate-related events and processes of most relevance to the Olkiluoto area are:

Isostatic rebound – the Olkiluoto area is still recovering from the last glaciation (Weichselian), which ended about 10 000 years ago. The ongoing isostatic rebound that followed this glaciation has major implications for the evolution of surface hydrology and for sea-land relationships. This is being taken into account in landscape modelling and in hydrogeological models.

Permafrost and ice-sheets – current understanding of the possible development of permafrost and ice-sheets will be updated, taking into account the most recent relevant literature available on the subject. The consequences of permafrost and ice-sheet development on the repository and on surface conditions will be estimated using appropriate models. The data supporting this modelling work will be obtained from the site whenever possible. With respect to the possible geohydrological and hydrogeochemical changes that may be caused by ice-sheet and ice-sheet melting, the promise and feasibility of field studies in Greenland are currently being explored in a joint preparatory project by SKB and Posiva. The objectives of the field studies are:

- gaining information in ice-sheet hydrology,
- collecting temperature data at the ice melting margin,
- measuring the depth of permafrost at the western margin of the Greenland ice sheet,
- searching for deep fractures at the melting margin from which to collect groundwater and analysing its composition, so as to determine the chemical characteristics of any penetrating meltwater, and
- collecting data for landscape models (e.g. vegetation patterns).

Post-glacial earthquakes – large earthquake in the Olkiluoto area are most likely to occur after future glacial episodes, the timing of which will be provided by the climatic scenarios. The consequences of post-glacial earthquakes for the repository will be estimated in the analysis of relevant scenarios.

3.2 Site characterisation and overall understanding

Olkiluoto site characterisation activities have been ongoing for over 20 years and there is an increasing level of confidence in the Olkiluoto site understanding. Complementary site investigations are carried out within the *VARTU programme*, both from surface and from ONKALO. The monitoring programme at Olkiluoto (*OMO-programme*) has been established to monitor the effects of construction activities at the Olkiluoto site on the surface environment and on the mechanical, hydrogeological, and hydrogeochemical properties of the rock. Boreholes constructed both from the surface and from ONKALO are monitored. Results are reported annually by each discipline. The introduction of foreign materials to the site is also being tracked in this programme.

The *ONKALO underground rock characterisation programme* provides essential information that contributes to site understanding and to understanding of the disturbances due to excavations (e.g. on groundwater flow conditions). Demonstration activities are also planned at ONKALO at repository depth, although these plans are still under development.

The technical means needed for groundwater inflow management in ONKALO have been studied in the *R20 programme*, the final report of which is currently being finalised. These technical means must be efficient in keeping the inflows small, but at the same time must be acceptable from the long-term safety point of view. The programme was divided into three parts: grouting materials, grouting techniques and long-term safety aspects. On the basis of the programme, a new low-pH grout has been introduced, and improvements in the techniques for emplacing the grout have been made.

A special programme has been established to characterise, control and study the impact of the excavation damaged zone (EDZ) around the excavations. The programme is focussing on the excavation damaged zone around ONKALO. It is expected that the characterisation activities will bring input about the EDZ properties that can be used in safety assessment modelling.

Assessments of the impact of ONKALO, incorporating site data and using hydrogeological models, as well as observations made during the on-going excavations, have been reported in Löfman and Mészáros (2005), Ahokas *et al*. (2006) and Pastina & Hellä (2006). The disturbances have also been discussed within the *R20 programme*. Prediction-outcome studies are being performed at ONKALO as an integral part of the site characterisation to improve the understanding of the rock mass around ONKALO (see Chapter 9 of Andersson *et al.* 2007).

Prediction-Outcome studies are also performed in ONKALO as an integral part of the Site Description to improve the understanding of the rock mass around ONKALO (see Chapter 9 of Andersson *et al.* 2007). Demonstration activities are also planned in ONKALO at repository depth but plans are still under development.

The results of the site investigations are assessed and site descriptive modelling of the bedrock is carried out by the *Olkiluoto Modelling Task Force (OMTF).* The site descriptive model presents a synthesis of the geoscientific understanding gained by the site characterisation activities, as well as information on the processes affecting site evolution. The most recent site description has been published in 2007 (Andersson *et al.* 2007) and updates are planned for 2008 and 2010. These reports will provide key input to the *Models and Data Report* (see Section 2.4), and, along with background reports discussing the measurements of e.g. groundwater flow, will also provide as basis of the development of the EBS.

The *Olkiluoto Biosphere Description (BSD)* summarises the site data on the ecosystems, including the overburden, and scientific understanding of the biosphere and its linkage to the bedrock conditions, as studied by the OMTF. In the BSD, descriptive models of ecosystem types and their functions are presented for further application in the assessment. A large part of the data is produced within the *Olkiluoto (environmental) monitoring programme* (Posiva 2003c), including the relevant parts of the surveillance programmes of the nuclear power plant. The first version of the BSD report was published in 2007 (Haapanen *et al*. 2007) and an updated and complemented version will be published in 2009.

3.3 EBS design and development

An overview of the current stage of the EBS development was included in TKS-2006 (Posiva 2006). The development of the EBS is currently ongoing, with the aim of defining feasible performance targets for the main barriers by the end of 2009.

Posiva has identified a reference method for canister fabrication. According to this method, the copper overpack, with an integral base, is manufactured by a pierce-and-draw technique. The copper lids are manufactured by a hot-pressing technique (which is sometimes called "forging"). Posiva plans to seal the lids of the copper overpacks with electron beam welding (friction stir welding is being investigated as an alternative).

A quality assurance and quality control programme for canister manufacturing, conditioning and sealing is being developed, as described in Posiva (2006), in SKB (2004) and in Andersson *et al*. (2004). The reference sealing method for the construction license application will be decided within a couple of years based on the results from this quality assurance and control programme.

Preliminary canister manufacturing specifications, inspection programmes and inspection criteria will be included in the update of the Canister Design Report to be published by the end of 2008. Reports describing the canister inspection technology and manufacturing technologies, including the welding technology, are to be published by the end of 2009.

Buffer and backfill design activities are also ongoing. Design reports will be produced by the end of 2009. Several programmes have recently been established to better address important issues related to the buffer and backfill performance, and the interaction between long-term safety issues, design requirements and the site descriptive model.

The *BENTO-programme* has been established to study the properties of bentonite and related processes through both experimental and modelling studies. The objectives of the programme are to:
- develop both theoretical and experimental knowledge of THMC processes in bentonite,
- specify the conditions and bentonite properties needed for the bentonite components to fulfil their functional requirements in the EBS,
- study different bentonite materials in conditions representative of those in a repository, and produce starting points for the design of bentonite components, which are a prerequisite for the evaluation of the performance of the buffer,
- build up sufficient national expertise and resources for the design of bentonite components and for demonstrating the required performance of the bentonite buffer for the PSAR needed in support of an application for a construction license,

- generate the expertise needed to perform these RTD activities, making use of existing experts, but also educating new experts in those research areas identified as being critical.

Posiva is currently developing alternative backfilling concepts for the deposition tunnels and for other excavated parts of the repository (transport tunnels, auxiliary rooms, access tunnel, shafts) as well as methods for the final closure of the repository, such as plugging and sealing methods. The backfilling and sealing concepts have been developed together with SKB within the Baclo (Backfilling and closure of the repository) programme. A report on backfill methods and plans has been published recently in collaboration with SKB (Gunnarson *et al.* 2007) and will be updated based on the information gained in the last phase of Baclo (phase III). This report will discuss how different combinations of materials and methods fulfil the requirements set for the backfill. The *Coordination programme for the Olkiluoto-specific backfill (OBA)* has been established to develop a site-specific backfill design. Olkiluoto-specific issues related to design and long-term safety will be identified, and backfill materials selected, consistent with the concept for the deposition tunnels and for the Olkiluoto repository as a whole. The OBA programme is divided into the following tasks:

- development of the Olkiluoto-specific backfill design,
- development long-term safety requirements for the backfill,
- selection of backfill materials, characterisation of material properties and their influence on the design specifications,
- cost calculations and comparisons, and
- production of an Olkiluoto-specific backfill design report, which summarises the knowledge acquired within OBA.

The Olkiluoto-specific backfill design report will be published before the end of 2009.

3.4 Repository design, construction and operation

Repository design and plans for construction and operation are ongoing, with the site descriptive models providing essential input, particularly for the repository layout adaptation.

In the case of KBS-3V, a stage 2 report detailing a preliminary repository design has been compiled by Saanio *et al.* (2006). The current outline planning stage aims to develop a facility description based on this concept by the end of 2009. In the case of KBS-3H, a preliminary repository design report for a KBS-3H type repository has been compiled by Autio *et al.* (2008). A complementary study stage extending over the period 2008 - 2010 will undertake further development of the design. Examples of layout adaptation for a KBS-3V type repository at Olkiluoto have recently been prepared by Kirkkomäki (2007) and, for a KBS-3H-type repository, by Johansson *et al.* (2007).
A description of KBS-3V repository construction and operation activities (including encapsulation) has been reported in the Facility Description 2006 (Tanskanen 2007). Planned activities related to repository design, construction and operation for the period

up to 2009 are described in TKS-2009. Closure concepts for deep repositories are being developed in cooperation with SKB in the Baclo programme (see Section 3.3).

3.5 Requirements management

Posiva´s requirements management system, *VAHA,* has been established to define, document and manage the requirements of different origin (e.g. regulatory, engineering, stakeholders) set on different components of the repository EBS. Performance targets (see Section 2.1.2) for the system components, e.g. canister, buffer, host rock, will be specified in the VAHA database with the appropriate connections between the different requirements. The *VAHA system* thus includes, for example, design specifications and information related to the range of conditions under which the system will perform as required.

The Rock Suitability Criteria (RSC-) programme has been set up to define the performance targets for the host rock and to develop the criteria for accepting certain rock volumes for disposal, including the acceptance criteria for the deposition holes. The criteria to be applied should cover the requirements arising from both long-term safety and design. As a result of applying the criteria, estimates of the expected conditions around the deposition holes will be achieved, along with the probabilities of deviations that may lead to disruptive events. Examples include estimates of flow conditions around deposition holes and input to the estimation of the likelihood of canister failure by rock shear in the event of a large earthquake.

4 MANAGEMENT OF UNCERTAINTIES

4.1 General principles

The overall strategy for the management of uncertainties can be summarised in four words: **identify**, **avoid**, **reduce** and **assess**. **Identification** and communication of uncertainties are usually 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 **avoiding** concepts and components the behaviour of which would be difficult to understand and predict. The stepwise implementation process of the repository allows the **reduction** of uncertainties by means of continuous RTD efforts. However, some uncertainties will always remain and have to be **assessed** in terms of their relevance to the final conclusions on safety.

According to the Safety Case Plan 2005 (Vieno & Ikonen 2005), the assessment of uncertainties will be based on a mixed approach, in which probabilistic methods will be used to describe variability whenever practicable, whereas the epistemic uncertainties will mainly be handled by analysing their impact on system performance through a range of deterministically defined calculation cases in safety assessment. The models used for both types of analyses will include assumptions that are conservative which means that they shall ensure that "*the results of the safety analysis, with high degree of certainty, overestimate the radiation exposure or radioactive release likely to occur*" (STUK 2001 YVL 8.4). The deterministic analyses will include bounding calculation cases that explore extreme situations. Sensitivity analysis will also form an essential part of the assessment.

In its review of TKS-2006, STUK considered Posiva's approach to be in compliance with STUK's recommendations, but it requested a more explicit description of the practical plans to manage uncertainties (STUK 2007).

The Safety Case Team that supported STUK in the review arrived at a different conclusion: according to their impression, Posiva does not attempt *"to satisfy the requirement for formal uncertainty analysis"*. According to the team, *"uncertainties in knowledge and data need to be tracked and turned into suitable probability distributions"*. For that purpose *"tools suitable for uncertainty analysis are readily available, but without a proper commitment to doing this, the thinking required to define suitable probability distributions for parameters, including correlations, will not take place"* (STUK 2007).

Somewhat similarly, in the recent review of Posiva's Olkiluoto Site Description 2006 (Andersson *et al.* 2007) a strong call for quantification is expressed: *"The report discusses the uncertainties of the results and models in different disciplines, but only qualitatively. Posiva shall assess uncertainties also in a quantitative way"* (STUK 2008). Analogous comments were made in STUK´s review of the Evolution Report (Pastina & Hellä 2006): "*with respect to justification of the normal evolution, uncertainties in processes or data are often stated but in most cases not quantified, so their safety consequences remain unclear*". (Apted *et al.* 2008).

In Posiva's view, most of the safety case is about discussion of uncertainties. The most important part of the management of uncertainties is the RTD that has been carried out since the late 1970's and will continue in the future. In this respect, the planning of Posiva's RTD programme as a whole is about managing the risks and uncertainties related to safe handling of spent nuclear fuel, as illustrated in Figure 4-1. Clearly, some uncertainties will always remain and they have to be assessed to support decision making. However, most of the residual uncertainties cannot be easily quantified.

Apart from some attempts to use fuzzy sets (NEA 2004b), the quantification of uncertainties in safety assessments has generally been based on probabilistic methods. In the late 1980's there was considerable interest in fully probabilistic safety assessment methods and codes, but these achieved only limited success. In fact, one of the main criticisms of the Post-closure Safety Assessment produced by the Canadian AECL in 1994 concerned the probabilistic approach adopted. The shortcomings of this and other similar attempts are well described, e.g., in the recent "European Pilot Study" by a number of European safety authorities (Vigfusson *et al.* 2007).

Nowadays, probabilistic methods are being used in different ways in safety assessments, and the problems and pitfalls of the methods are usually well understood. In many countries, regulations are based on risk criteria, which require probabilistic methods at some stage of the analysis. This is the case, e.g., in Sweden, although the value of deterministic analysis to illustrate the meaning of critical factors is also acknowledged there (Dverstorp, Strömberg *et al.* 2008).

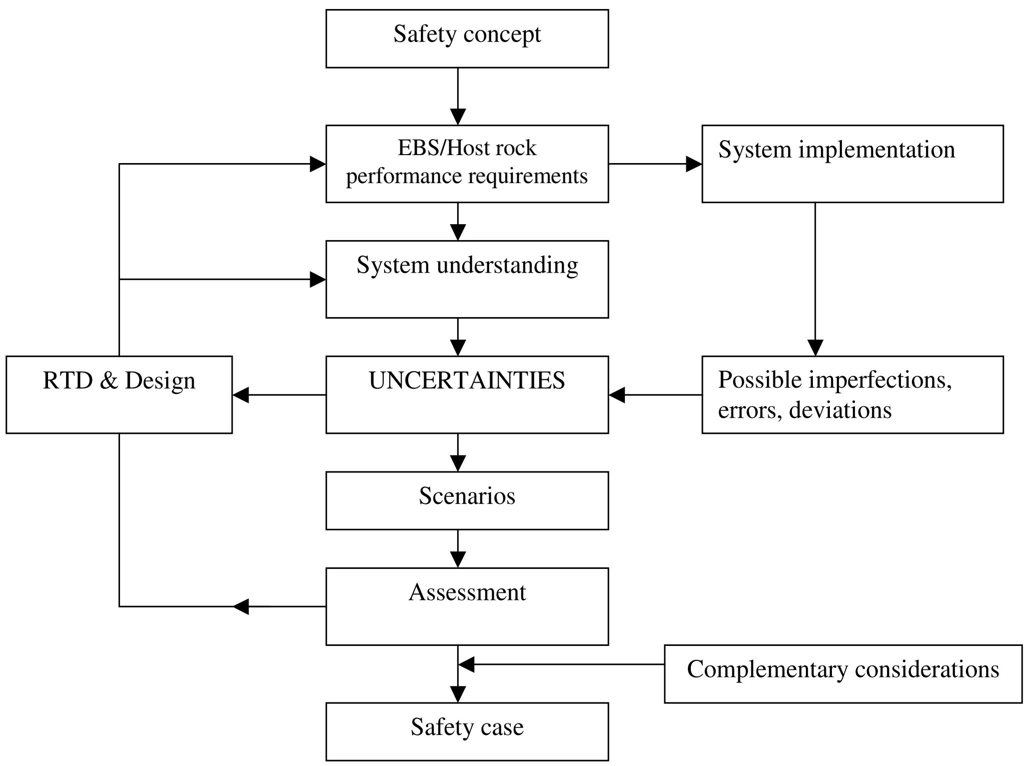

Figure 4-1. *Relationship among uncertainties, safety case and research and technical development (RTD) and design activities.*

In spite of this progress, opinions still differ on how uncertainties should best be handled in safety cases. In a report of an international expert team working for SKI and SSI in Sweden, the establishment of "a central register of uncertainties" was strongly recommended (Sagar *et al.* 2008). According to the report, the register should contain descriptions of each uncertainty, together with a *"thorough description of the methodological means of assessing [it] qualitatively and/or quantitatively"*, plus some background information. However, the Swedish authorities do not see any apparent benefits from such a register (Dverstorp, Strömberg *et al.* 2008). The reason, they say, is that uncertainties and their treatment are already the essence of any safety assessment report.

Posiva agrees with the Swedish authorities: the uncertainty analysis cannot be separated from the rest of the safety assessment without duplication of large parts of the argumentation. Quantification and mathematical treatment of uncertainties may be possible in some cases, but it does not necessarily make the assessment more exact than thorough verbal explanations. In fact, the quantification of uncertainty in complex matters or issues usually means losing information and transparency. For instance, most of the site descriptions consist of differentiating between what is known and unknown; the numbers could never convey the same meanings as these descriptions.

Following the basic approach explained in the Safety Case Plan 2005 (Vieno & Ikonen, 2005), Posiva is now striving towards a more systematic and comprehensive discussion of uncertainties and discussion of uncertainties is integrated in all reports. For example, the *Models and Data Report* will address data uncertainties explicitly. The use of probabilistic analyses will mainly be limited to the parts of the assessment in which some objective basis for the quantification of uncertainties in terms of probabilities exists. Deterministic calculations using parameter values selected from specified ranges of variation can be considered as special cases of probabilistic analysis, as they can be considered to represent uniform probability distributions. In fact, if only the upper and lower bounds to the range of variation is known for a parameter, the maximum entropy principle would call for using such a uniform distribution.

In other cases of uncertainty associated with random variability, the "objective" basis could consist of statistics or knowledge about the process that creates the randomness. The variability of material or site properties is often handled in statistical terms. However, in most cases, "true" statistics of the variability is lacking or insufficient and, therefore, the statistical analysis normally requires additional considerations, e.g., to acknowledge the heterogeneity properly. An understanding of the processes underlying the randomness can be used to support the choice of probability distributions, but it is seldom available to the extent needed. Therefore, even probability distributions based on some kind of statistics or process understanding include a component of epistemic uncertainty which cannot be estimated objectively.

Of course, the probabilities could also be based on pure judgments without any specific information on the underlying statistics or process understanding. However, as in the case of reactor safety assessments (Apostolakis 2005), the use of subjective probability judgments often leads to problems in the end-use of the results. The decision-makers (e.g., regulators) may have difficulty in accepting the results of such assessments unless there is a consensus on the distributions. Therefore, assessments based on subjective probability estimates are best used as personal decision aids if the decision-makers can directly feed their own probability estimates to the assessment system. In cases where the decision-maker has no control on the choice of the probability distributions, there may not be too much confidence in the findings of the assessment.

One challenge in the assessment of uncertainties is in dealing with correlations between parameters. It has been suggested that models might be formulated in such a way that parameters are independent of each other, and can thus be varied or sampled individually. However, to do so usually requires in-depth understanding of the process structure, and if that information were available, the correlation problem would normally not be there at all. According to NEA (2004b), up until 2004 at least, there had been no attempts to assign correlation structure to the probabilistic parameters in safety assessments.

4.2 The approach to uncertainty management

Posiva's approach to uncertainty management is largely in line with the recommendations of the European Pilot Study (Vigfusson *et al.* 2007) as described in the following steps.

4.2.1 Uncertainty in theoretical (conceptual) understanding of the FEPs

The assessment of theoretical or conceptual understanding of the features, events and processes is an integral part of Posiva's planning of RTD programmes and production of the main safety case reports and its supporting reports, such as the Process Report (Miller & Marcos 2007), the Evolution Report (Pastina & Hellä 2006), the Olkiluoto Site Description 2006 report (Andersson *et al.* 2007), the Biosphere Assessment report (Ikonen 2006) and the Radionuclide Release and Transport Report (Nykyri *et al*. 2008). In addition to informal discussion and argumentation, various systematic representations have been used: in the Site Description 2006, this has taken the form of tabulations as part of the overall consistency and confidence analysis; in the biosphere assessment, a pedigree analysis based on the ideas of Funtowicz & Ravetz (1990) is introduced. Similar methods will also be used in the forthcoming safety case reports.

Where there is inadequate knowledge or an insufficient basis for discriminating between different theories and theoretical assumptions, different calculation cases can be defined in safety assessment to represent and bound the ranges of possibilities.

4.2.2 Uncertainty in the models used to describe the processes

Assessment of model uncertainties is an integral part of all the reports presenting results of model simulations. This includes the consideration of uncertainties arising from the interdependencies of model assumptions and interactions between processes. The uncertainties in the models related to critical assumptions and data for safety assessment will be discussed in the *Models and Data Report.* Alternative model conceptualisations can be used to assess the impact of uncertainties on model outcomes, whenever such alternatives can be supported on theoretical grounds. Bounding calculation cases can be used to assess the consequences of extreme situations, although these should remain within the range of possibilities supported by current understanding.

4.2.3 Uncertainty in the data

The assessment of data uncertainties is a key part of the *Models and Data Report*. As directed in YVL Guide 8.4 (STUK 2001, Section 4.3), the reference data and parameters used in the model calculations will be based on conservative assumptions, but, whenever possible, the calculation cases based on best estimates will also be defined and analysed. The assessment of uncertainty in the *Models and Data Report* is mainly to determine the range of variation consistent with current understanding, including "pessimistic" values of the parameters (see Vigfusson *et al.* 2007). The uncertainties will be quantified by ranges of variation or by probability distributions to the extent that statistical or other reasonably objective data are available for this purpose. If there are insufficient data or scientific understanding to fully justify a range of variation or probability distribution for a parameter, stylistic approaches can be used whereby the consequences of different illustrative parameter values are evaluated.

The use of deterministic or probabilistic methods in the *Analysis of Scenarios Report* is discussed in Section 2.5.2.

Formal expert elicitation (see Section 5.6.1) will be applied in special cases for which data or theoretical understanding are insufficient or controversial and the assessment has to be based on judgements in coordination with safety assessors, as described in Section 2.4.4. As a normal procedure, experts specialised in a particular subject area will also provide their views on the uncertainty linked to a specific parameter.

4.2.4 System sensitivity to uncertainties in the data, parameters and alternative theoretical assumptions

Sensitivity assessment is used in general to examine 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 sources of uncertainty that have the biggest impact on the outcome of the analysis.

Two main tasks included in the sensitivity assessment can be identified: assessing the impact of different assumptions and input parameter values on the outcome, and assessing the impact of uncertainty in the input parameters on the uncertainty in the outcome.

The first task can be carried out only to the extent that suitable models are available to examine different assumptions and parameter values; otherwise the meaning of the uncertainty has to be studied by scoping the effect of various simplifying conservative assumptions. In the case of linear models, the sensitivity of the results to the input can be tested separately for each parameter by a series of calculations around a fixed base case. After exclusion of physically impossible or highly unlikely combinations, sets of assumptions can be defined for further analysis. In the case of non-linear relationships, a systematic analysis of the sensitivity to various parameter combinations and a more sophisticated sampling system is needed to identify conservative parameter combinations. Multivariate techniques may be useful for such system sensitivity assessment. Otherwise, the sensitivity to parameter combinations will be based on systematic sampling, using, e.g., Monte-Carlo methods or techniques from the design of experiments. For screening purposes, simplified models are used.

The second task is addressed by applying global sensitivity analysis, which studies how the uncertainty in the output of a model can be apportioned to different sources of uncertainty in the model input. Sensitivity analysis is, for example, useful for identification of parameters or processes for which increased RTD effort is justified for decreasing uncertainties, or increasing process understanding. Such sensitivity analyses have been performed recently for radionuclide transport models in the biosphere (Broed 2007), by means of a code called EIKOS, which has been developed and tested to facilitate such analysis (Ekström & Broed 2006). The use of EIKOS for other parts of the radionuclide release and transport assessment is being considered.

4.2.5 Range of uncertainty in results

The assessment of the total combined uncertainty is focused on the differences between the results from different scenarios and calculation cases and the ranges of uncertainty within the results for each case. A basic division is made between the expected evolution scenarios, for which the probability of occurrence is considered to be close to one, and the scenarios representing disruptive events. For the latter, the probability of occurrence will generally be small, but may be difficult to quantify. For example, in the case of human intrusion scenarios mentioned in YVL 8.4 (e.g. core drilling), it is not possible to assess the probability of such events because human behaviour cannot be predicted within tolerable levels of uncertainty. Therefore, argumentation about the likelihood of the scenarios will be based mostly on qualitative reasoning, except in those cases where it is possible to assign probabilities to disruptive events. An example of the latter is the probability that a canister will be damaged by rock shear movement occurring as the result of a large earthquake, which was estimated quantitatively in SR-Can (SKB 2006a).

For each scenario, the quantification of total uncertainty in the calculated consequences is based on deterministic analyses of the effects of those parameter uncertainties and theoretical assumptions that are found to be important on the basis of sensitivity analysis. The main purpose is to obtain the limiting estimates of the worst case consequences, represented by pessimistic parameter values or combinations. For these deterministic analyses, input presented originally in probabilistic form is first converted to best, conservative and pessimistic values using appropriate percentiles of the distributions.

Finally, the assessment of the total uncertainty involves the discussion of the residual uncertainties arising from matters that are beyond any evaluation. Scenario completeness is one such consideration. Others include the possibility of human error in the planning and implementation of the repository and the impact of future human behaviour on the repository (i.e. the possibility of human intrusion) and on estimates of radiological consequences. Completeness can never be proven. Measures can, however, be taken to promote scenario completeness, such as the comparison of the FEPs considered with international FEP databases or databases compiled and applied in safety assessments carried out by other national programmes (see Section 2.3). Bounding analyses and physical constraints can also help in confining residual uncertainties. The possibility of human errors affects all stages of repository implementation. It is also inherent in the safety case activities. Measures will, however, be taken to minimise the possibility of such errors, and these measures will be reported as part of the safety case. The unpredictability of future human behaviour adds uncertainty to the estimates of radiological consequences that cannot be expressed in quantitative terms. The likelihood of human intrusion scenarios can, however, be discussed in qualitative terms and argued to be low.

4.2.6 Potential to diminish the uncertainties

As mentioned above, some of the uncertainties cannot be reduced (e.g. human behaviour in the future) and some have a negligible impact on the safety case so they do not need to be reduced. However, for the residual uncertainties that have an impact on the safety case, the most efficient means to manage them is through RTD and design efforts and efficient management of the quality of the implementation. The assessment of further RTD needs and opportunities is an essential part of the safety case through all its phases and component activities as shown in Figure 4-1.

5 MANAGEMENT OF QUALITY

5.1 Goals and principles

Posiva applies a management system based on the ISO 9001:2000 standard for all activities, including the production of the safety case reports, and requires the pursuit of the same quality assurance principles from all its contractors and suppliers. The system was first launched in 1997 and has later been subject to continuous maintenance, updating and several internal and external audits. Regarding the activities related to ONKALO, the management system also takes into account the requirements of YVL Guide 1.4.

The Safety Case Plan 2005 was implemented as a project with customary project organisation, control and supervision systems as defined in Posiva's management system and documented in the “SAFCA” project plan. According to the project plan, the responsibility for the project as a whole has been with the project manager, and the responsibilities for the portfolio reports have been assigned to the project group members. Regarding the management of quality, the project plan refers to Posiva's general management system and the procedures and instructions defined therein.

The purpose of Posiva's management system is to ensure, in a documented and traceable way, that Posiva's products - whether in the form of abstract knowledge and information, published reports or physical objects - fulfil the requirements set for them. The general quality objectives, requirements and instructions defined in Posiva's management system will also form the foundation for the quality management of safety case activities carried out in the future. However, special attention will be paid to the management of the processes that are applied to produce the safety case and its basis. The purpose of this enhanced process control is to offer full traceability and transparency of the data, assumptions, modelling and calculations.

While the approach is based on the ISO 9001:2000 -type management, which means management through processes, the principle of a graded approach, as proposed in the safety guides for nuclear facilities, is pursued in the safety case production. The graded approach means that the primary emphasis in the quality control and assurance of safety case activities is placed on those parts of the assessment that have direct bearing on the arguments and conclusions on the long-term safety of disposal (marked in blue in Figure 1-6), while standard quality measures may be applied in the supporting work.

The *Models and Data Report* will act as an interface between the safety case activities and the principal supporting activities: the information included in the *Models and Data Report* is selected on the basis of its safety relevance: the EBS and site data that directly provide the input to the safety case are discussed in these reports, while more details can be found in the supporting background reports, such as the Site Descriptive Model report and various technical descriptions of the engineered barrier system. The quality of the Site Descriptive Model report is mainly ensured by the application of scientific principles, while the methods of quality control for the technical barrier design and implementation depend on the nature of the materials and technology in question. The

Models and Data Report captures the most significant information related to safety, the quality of which is of primary importance for confidence in long-term safety.

5.2 Safety case process

The safety case production will be based on the iterative process shown in Figure 1-7. The production process is divided into four main sub-processes:
- conceptualisation & methodology,
- handling of critical data and modelling,
- safety assessment, and
- evaluation of compliance and confidence.

The *conceptualisation & methodology* sub-process frames the assessment providing the description of the disposal system, FEPs and the formulation of scenarios, including the system's evolution. It guides the definition of the performance targets for the EBS and the bedrock, which will form the core of the requirements management system (VAHA). The VAHA system is currently under development at Posiva. The rock suitability criteria (RSC) programme has been established to develop and test the suitability criteria for the Olkiluoto bedrock. The VAHA and the RSC programmes are discussed further in Section 3.5.

The EBS performance requirements will guide the work on the *design and development of the Engineered Barrier System* that delivers the actual technical repository system and its implementation plan. Likewise, the ongoing programme on the *rock suitability criteria* will establish the goals and directions for the site investigations and modelling activities in such a way that the critical data needed for safety assessment can be obtained.

The *handling of critical data and modelling* sub-process makes the central linkage between Posiva's main technical and scientific activities and the production of the safety case. It can be considered as a clearinghouse activity between the supply of, and demand for, quality-assured data for the safety assessment. The data are produced by Posiva´s planning, design, and development processes for the EBS, by the site characterisation process for the geoscientific data and through the biosphere description of the Olkiluoto area. The Safety Case process is responsible for stating the most significant data demands as exactly as possible.

The *safety assessment* sub-process analyses the consequences of the evolution of the repository in various scenarios, classified either as part of the expected evolution or as disruptive scenarios. It is also responsible for assessing the environmental and human health consequences of radionuclide releases in cases where the canister no longer provides complete containment.

The *evaluation of compliance & confidence* sub-process is responsible for the final evaluation of compliance of the assessment results with the regulatory criteria and the overall confidence in the safety case, taking into account the completeness of the scenarios considered, uncertainties within the assessment and complementary considerations regarding the long-term safety of geological disposal.

Each sub-process can include specific tasks that will be carried out during the safety case production process as described under Safety Case Organisation (Section 5.3).

5.3 Organisation

The implementation of the safety case production process is based on the organisational structure shown in Figure 5-1. The process is itself a part of Posiva's core process, which aims to submit the construction license application at the end of 2012 and is owned by Posiva's Research Department (customer representative in Figure 5-1). Similarly, an owner is designated to each of the sub-processes described in Section 5.2. A special *Core Group* is set up to manage and coordinate the resources and time tables and assist in controlling the overall process of safety case production in the framework of Posiva's management system. The Core Group oversees three tasks within different sub-processes: 1) Data handling, 2) Conceptualisation and Methodology, and 3) Safety Assessment. The tasks will be carried out by members of the safety case group supported by subject-matter experts.

A quality manager function is designated for the quality assurance activities. This includes the regular auditing of the process, review and approval of the products and the management of expert elicitation (Section 5.6.1).

A steering group will be nominated to supervise the progress of the work from the perspective of Posiva's overall goals and constraints.

The organisation of the safety case production process is referred to as SAFCA.

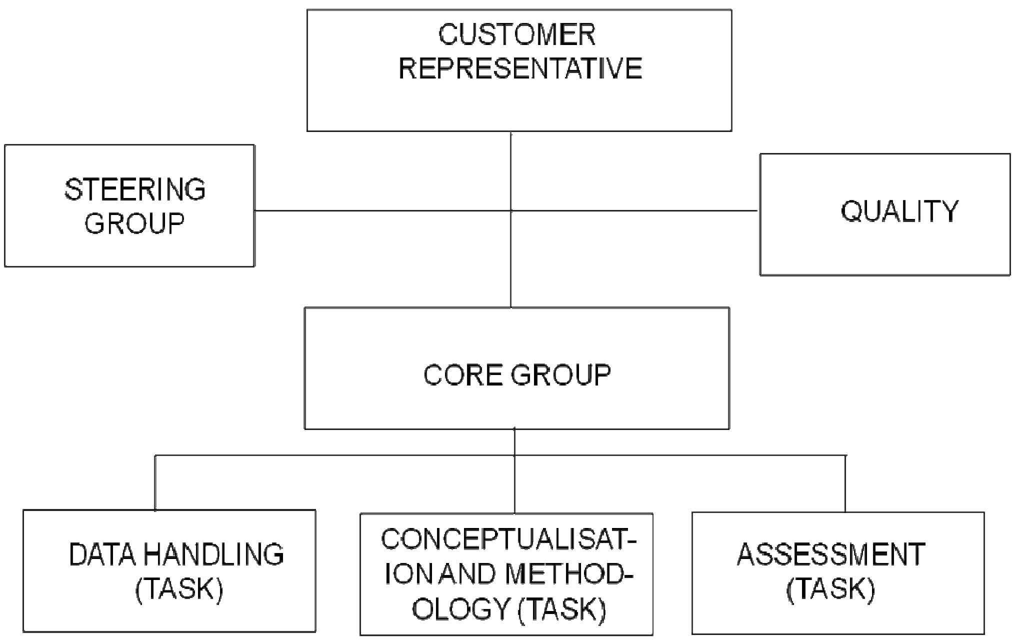

Figure 5-1. *Organisation of the safety case production process (SAFCA).*

5.4 Quality management of the safety case process

5.4.1 Main customer process

The purpose of the safety case process is to produce the arguments/documentation on long-term safety needed for the licensing of the repository. The overall control and supervision of the safety case process is handled through a customer process that consists of:

- managing the customer interface:
  - agreement on the goals and constraints of the work
  - control of costs and time schedules
  - allocation of personal resources
  - identification of the needs for additional RTD efforts
  - management of change
  - verification of product compliance with goals and constraints
- controlling the production processes:
  - assignment of sub-process tasks
  - control of sub-process interfaces
    - matching the input/output requirements within the process
    - matching the time schedules
  - maintenance of internal compliance and consistency
  - management of non-conformities.

The main goals and constraints of the work are defined in the present Safety Case Plan, which will be updated by the Core Group as the need arises. For the purposes of cost control, annual and three-yearly budget plans are prepared by the Core Group and approved according to the rules defined in Posiva's management system. The Core Group follows the current balance of the budget and reports regularly on the progress made to the owner of the safety case process. The allocation of resources is a part of the annual planning and budgeting discussion, as is the need for additional RTD efforts.

The Steering Group convenes regularly to follow the progress of the work and reviews the products of the work in relation to the stated objectives. It approves the proposed changes in the Safety Case Plan and assesses the needs for enhanced RTD activities. The actual decisions on RTD activities are made according to the responsibilities defined in Posiva's management system.

On the basis of the Safety Case Plan, the Core Group:

- prepares and maintains an activity plan that defines the inputs and outputs expected from the safety case sub-processes,
- coordinates the execution and time schedules of the actions arising from the activity plan,
- oversees the consistency of data and assumptions used in the various tasks of the safety case process and sub-processes, and
- takes appropriate actions in case of non-conformities.

All non-conformities that have a bearing on the substance of the published safety case documents are reported.

5.4.2 Conceptualisation and methodology

The framework for data collection and for assessment activities is defined by the *conceptualisation and methodology* sub-process. It

- maintains a current definition of the disposal system and its environment for the purpose of safety case activities (documented in a safety case portfolio report),
- maintains the database of features, events and processes (documented in a supporting report),
- develops the general methodology for formulation scenarios and applies this methodology to define the scenarios to be analysed in safety assessment (documented in a safety case portfolio report),
- coordinates the management of uncertainty throughout the whole safety case process (taking into account the theoretical and conceptual uncertainties considered in the FEPs reports and in the formulation of scenarios, the data uncertainties in the assessment of data and models, the treatment of uncertainties in the analysis of scenarios and the overall combined uncertainty in drawing the conclusions) - the actions taken are documented in the activity plan, and
- feeds the conclusions of the safety case to the VAHA process (requirements management system) and the RSC process (rock suitability criteria): this comprises the assessment of the compliance with the current requirements and the observations of any non-compliance. All observed cases of non-compliance are reported in writing.

5.4.3 Data handling and modelling

The *handling of critical data and modelling* sub-process creates the main link between the *safety case* process, the *engineering design and planning of implementation* processes and the *site characterisation* process. It starts from the definition of the primary input for the safety case (the "critical" data, models and assumptions used in the safety assessment) and sees to it that all the steps needed to produce that input are described in a way that enables the control and assessment of its quality.

The sub-process consists of the following activities:

<u>1. Identification of the critical input data</u>

The first step is to identify the parameters and data sets that are needed as input to the safety assessment (Section 2.4). Identification is based on prior safety assessments and the needs defined by the *safety assessment* sub-process. However, the final definitions of the actual parameters and data sets are formulated as a clearinghouse activity in liaison with the supply processes (engineered barrier system design and development; site characterisation): the clearing house activity ensures that the input requirements are defined in a way that makes it possible for the supply processes to produce the input within the required time frames. A document containing the current list of data, parameters and models considered as critical input is prepared and maintained.

2. Description of the production processes for the assessment input data

The chain of activities that is needed to produce the input in the form needed by the safety assessment is described and documented. In most cases, this means the specification of the initial EBS and site conditions at the time when the control of the emplacement process ceases (e.g., after the emplacement of the buffer). For the site data, the chain of production activities starts with observations and measurements, continues with primary interpretations and necessary abstractions (inference, upscaling, expert judgement) and ends with the derivation of effective parameters or datasets that can be used in safety assessment models and calculations. Models and theories are used for interpretation and abstraction. For the EBS data, the production process is divided into two steps: the design stage and the implementation stage. The production process for assessment input data takes account of the reasoning behind the design requirements that led to the choice of the design parameters (specification). It also considers design justifications in terms of performance studies and practical demonstrations that show how the specification meets these requirements. Repository implementation procedures are relevant to the extent that they may affect the actual "as-built" values of the parameters and their associated uncertainties.

3.Quality review of the safety assessment input data

The description of the production processes for the most significant assessment input data (Step 2) is needed for their quality review. The responsibilities for the definition and implementation of the quality control and assurance measures for the supporting processes that produce the input data are defined within these supporting processes themselves. The quality review at this step consists of checking that a continuous chain of quality control has been applied to the production of the input and the prescribed quality assurance measures have been taken. The results of the review are summarised in the *Models and Data Report*, together with the references on which the review is based.

The input provided by expert elicitation, process is specified in Section 5.6.1.

4. Clearance

At this step, the collected input data is either validated for further use in the safety case, further proofs and justifications are requested from the data suppliers, or the data is released for restricted use. All restrictions and caveats on the use of input data are documented in the *Models and Data Report*.

5. Improvement

Based on the quality assessment of the input data, proposals for improvement of the quality management in different parts of Posiva's programme are made.

5.4.4 Assessment

The *safety assessment* sub-process carries out the actual analysis of scenarios and produces the estimates of the long-term impact of the disposal system on the human health and the environment in terms of indicators specified in the General Safety Requirements (STUK 1999) and YVL Guide 8.4 (STUK 2001). It is divided into two branches, as suggested by the same Guide: (1) the analysis of the expected evolution of the repository and (2) the analysis of the impact of unlikely disruptive events on repository safety.

The quality control and assurance of the assessment consist of the following steps:

- 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,
- verification of assessment codes,
- validation of the assessment codes for the intended application,
- documentation of input for the production runs,
- application of a procedure to ensure codes are correctly applied,
- documentation of the code versions used, and
- reporting of non-conformities.

Validation of models and data is based on the clearance and caveats on their use provided by the *handling of critical data and modelling* sub-process, but also includes the conformity with any requirements arising from special scenario and model assumptions.

5.4.5 Compliance and confidence

The purpose of evaluation of *compliance and confidence* sub-process is to produce reliable evidence, arguments and analyses that support the compliance of the intended disposal plans with the applicable safety requirements. The responsibility of the Core Group is to follow the progress made in the assessment regarding

- the compliance of the estimated environmental and human health consequences with the regulations, and
- the confidence in the estimates made.

Any point of non-compliance is reported to the Customer Representative, to the person in charge of the related requirements in the VAHA system, and to the Steering Group, which may propose additional checking and evaluations within the framework of the safety case process. Otherwise, the responsibility for further actions, e.g., new RTD activities, is transferred to Posiva's research and technical development (TKS) Group, as defined by Posiva's management system.

The evaluation of the reliability of the assessment and confidence in the conclusions is based on expert reviews that are made throughout the process (Section 5.6.2). The comments received from reviewers and the actions taken in the safety case process are both documented. The evaluation of confidence in the final licensing documentation is part of the Summary Report of the safety case portfolio. This evaluation also takes into account the complementary evaluations that are made outside the *analysis of scenarios.*

5.5 Quality management of supporting processes

As shown in Figure 1-7, the safety case production process relies on the input from two principal Posiva processes: the *site characterisation process* and the *engineering development and design process*. In addition, it provides the necessary input for the *requirements management process* and the *rock suitability criteria (RSC) programme*. The same system also applies to the biosphere characterisation.

The quality management of site investigations and the processing of the primary data from measurements and observations is based on the ISO 9000:2000 standard and covered by Posiva's management system. The system also covers aspects of the subsequent modelling and interpretation activities, including the responsibilities of the suppliers, supplier audits, application of computer codes, and use of data sets, documentation and product control. However, the quality achieved also relies on the application of scientific principles, such as the critical use of data and information, open publication policy, repeatability of the experiments and traceability of the information used.

The quality management of the *engineering development and design process* is based on system-specific quality manuals as will be the quality management of implementation. Such manuals will be produced for all the main components of the engineered barrier system: the canister, the buffer and the backfill and the sealing systems.

5.6 Special quality management function

5.6.1 Expert elicitation

The purpose of the *handling of critical data and modelling* sub-process is to describe the whole chain of activities that are needed to produce the input data and models for safety assessment. Ideally, the description should provide a logical and theoretically sound derivation of the input data and models from empirical observations, measurements and experiments. In some cases, however, the empirical or theoretical basis provided by experts in a relevant area is insufficient for discriminating between different models and assumptions that could potentially be used in the safety assessment. There may be conflicting data sources, alternative theoretical assumptions or simply a lack of relevant data. In such cases, the input data and models are selected on the basis of expert judgement. To make the judgements as transparent and unbiased as possible, a formal expert elicitation process will be introduced as a part of the data production.

The process is based on the interaction and collaboration between safety analysts and domain experts and aims at developing a joint understanding of the issue at hand and, consequently, a consensus view on the input data and models to be used in the assessment. The background and details of how this process could be implemented are described in a working report (Hukki 2008).

The process may consist of the following phases:

- selection of the issue,
- selection of the forum of elicitation,
- selection of experts,
- selection of the shared conceptual frameworks,
- preparatory work of safety analysts,
- training of experts,
- instruction of experts,
- independent work of experts,
- iterations,
- validation of expert judgments for later use,
- treatment of possible controversies, and
- final documentation of the process.

The process would be initiated and controlled by the SAFCA Quality Manager (see Section 5.6.3), who also would nominate the experts after consultation with the SAFCA Core Group. The way the process is arranged depends on the selected issue. For some issues, only a few experts may be available to provide input, while in other cases numerous experts from different disciplines may be available. In the latter case, the number of experts ideally needed for the elicitation and validation process depends on the degree of uncertainty and controversy surrounding the topic. Similarly, the process can be carried out in different fora. In the simplest cases, the elicitation may be carried out through exchange of questionnaires, but, since the most essential part of the process is the discussion between the participants, meetings or workshops will normally be preferred.

While it is important that the participating experts are highly qualified and that their expertise covers the issue at hand, it is not realistic to set formal rules for their selection. The basis of selection of the experts, as well as the qualifications of the experts selected, will be fully documented.

Conceptual tools will be used to facilitate joint understanding of the issue. There are different types of tools: *Elicitation Forms* for each party (safety analysts and domain experts) and *Structural and Contextual Descriptions* that serve as shared conceptual frameworks for both parties. The safety analysts formulate a preliminary view of input data and model uncertainties and document this view and the underlying assumptions and grounds in a *Safety Analysts' Elicitation Form*. The form guides the analysts in formulating, justifying and documenting their view in an explicit way.

The function of the *Experts' Elicitation Form* is to support the experts in the production of expert judgments.

After being informed by the safety analysts on the course of action foreseen for the elicitation and validation process, the experts are provided with an explanation of the purpose and idea of the form. The experts are instructed on what is required from them and on the format in which the judgment(s) must be presented. The instructions focus on how to address uncertainty and on the importance of making the reasoning process as explicit as possible. In the case of probabilistic judgments, special training may be useful.

Structural descriptions are intended to build a common understanding of the issue at hand and a comparison of the safety analysts' and the experts' views in a structured way. The descriptions provide shared frames of reference for describing and discussing uncertainties and possible inter-relationships between different uncertainties. Contextual descriptions are prepared to present the context of the production of the input and the intended use of the input in the safety assessment.

What follows is the provision of the requested input and subsequent discussions and iterations, with the aim of reaching a sufficient consensus to provide recommendations on the use of input data and models in safety assessment. If the safety analysts and the experts are unable to reach sufficient consensus, the resolution of the issue is left with Posiva's Safety Committee. The whole process will be fully documented.

In general, the functions and responsibilities for the process are assigned according to Table 5-1. Here "X" means major contribution, "x" a supportive contribution, and "[x]" means optional supportive contribution.

5.6.2 Expert reviews

All the reports of the safety case portfolio will be subject to expert review before publishing and special reviews may also be requested for intermediate progress reports. In particular, expert reviews will be carried out of all reports that do not carry disclaimers of Posiva responsibility. These include reports such as the Olkiluoto site descriptions and the main reports on EBS performance.

To ensure consistency, a special expert panel will be nominated to review the final portfolio reports that are intended to support the license application. The outcome of the reviews will be documented, together with the actions taken in response to the review comments.

***Table 5-1**. Roles and responsibilities of participants in the expert elicitation process. "X" means major contribution, "x" a supportive contribution, and "[x]" means optional supportive contribution.*

Phase	**Quality Manager**	**Safety analysts**	**Domain experts**	**Safety Committee**
Selection of issue	x	X		
Selection of forum	X	X		
Selection of experts	X	x		
Selection of shared conceptual frameworks	X	x	[x]	
Preparatory work of safety analysts	x	X		
Training of experts	x	X	X	
Instruction of experts	x	X	X	
Independent work of experts	x		X	
Iterations	x	x	X	
Validation of expert judgments for later use	x	X	X	
Treatment of possible controversies	x	X	X	X
Final documentation of the process	X			

*) (X = major contribution, x = supportive contribution, [x] = optional supportive contribution)

5.6.3 Quality manager

A quality manager will be nominated for all SAFCA activities. The duties of the manager include supervision of the quality management by regular audits and spot checks, clearance of non-conformities and general advice on application of the quality measures. The quality manager also nominates the experts for the reviews of the reports and has overall responsibility for ensuring that the expert elicitation processes is carried out according to the formal principles established.

6 IMPLEMENTATION

The shift from the 2005 Safety Case Plan to the revised structure and organisation will take place in the summer 2008. Accordingly, the Radionuclide Release and Transport Report for a KBS-3V repository (Nykyri *et al.* 2008) will be published as planned (in July 2008), but a new time schedule will apply to reports published thereafter. The first versions of the *Description of the Disposal System, Models and Data, and Summary Reports* will be published by the end of 2009. The schedule for the portfolio reports is shown in Figure 6-1.

The safety case will be a part of the documentation needed for the application for the construction license and the summary report will feed directly to the Preliminary Safety Analysis Report (PSAR; see Figure 6-2). Assuming that the construction of the repository can start during 2014, the Final Safety Assessment Report (FSAR) for the operational license application would be prepared by the end of 2018. A revised safety case is needed for that purpose.

The organisation of the work is discussed in Chapter 5. After adoption of the revised Safety Case Plan, Posiva's Management Group will nominate the Steering Group, who will approve the detailed work plans and assign the main responsibilities, including the Quality Manager, the Core Group members and the leaders of the Task Groups. The Task Group leaders will make a proposal regarding the nomination of Task Group members for approval by the Steering Group.

The Steering Group will follow the progress of the work and decide on possible changes needed in the work plans. In case of substantial revisions to the plans, a new version of the Safety Case Plan will be produced.

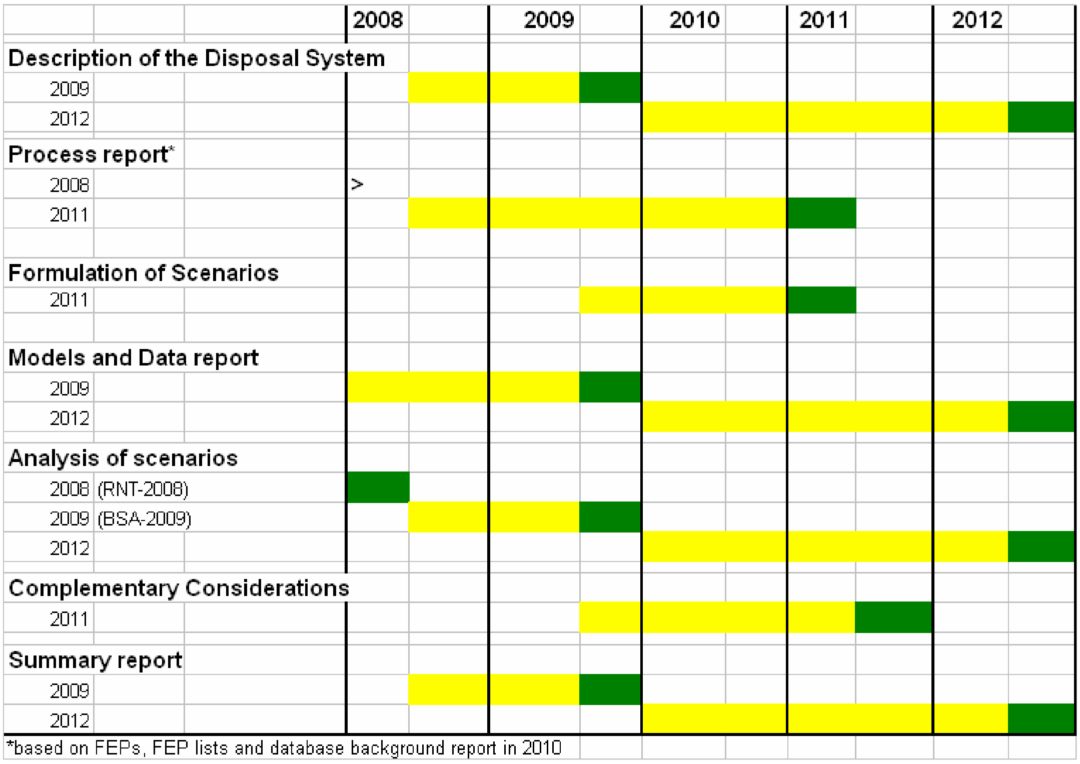

Figure 6-1. *The time schedule of the main reports in the safety case portfolio. The yellow colour means intensive work, draft versions and continuous updating of the reports. The green colour means final reports. (RNT 2008, Radionuclide Release and Transport Report to be published in 2008, BSA = Biosphere Assessment Report to be published in 2009). Recently published Process Report is noted with >.*

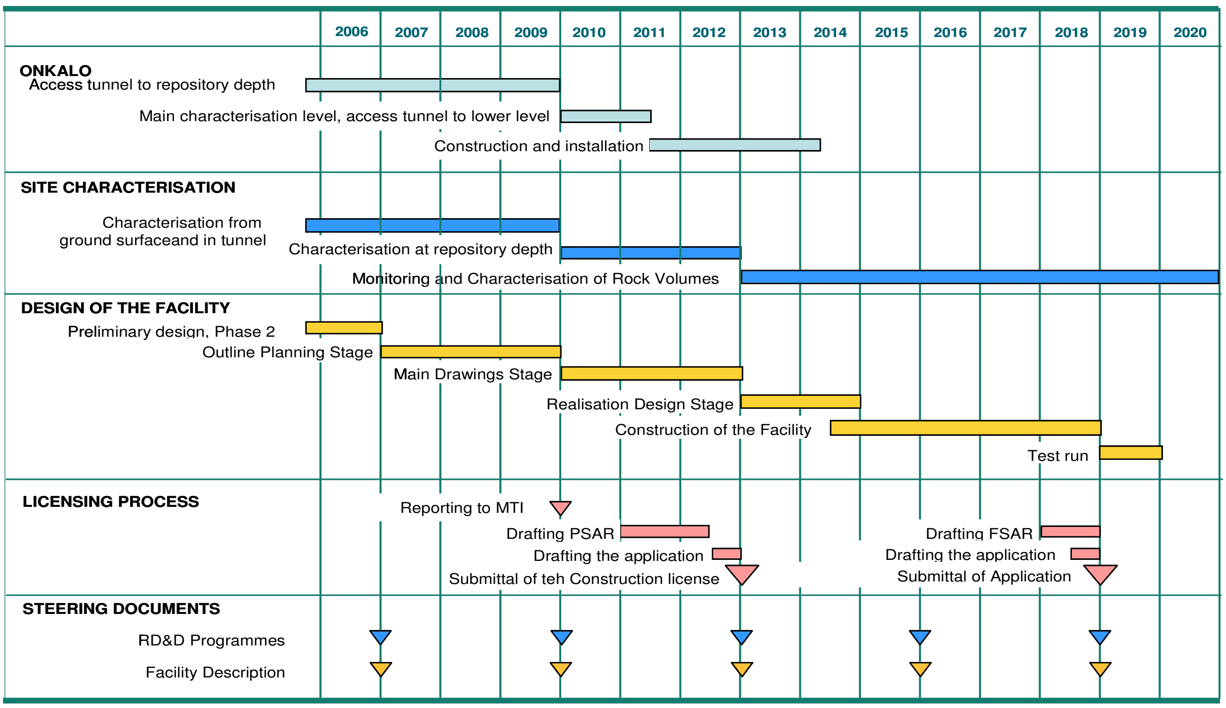

Figure 6-2. *Posiva´s overall timetable in view of the license application.*

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