TURMET - Turvallisuusperustelun metodiikan systematisointi

Literature Review on Safety Case in Nuclear Waste Deposition

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This report is on a literature review made as a part of the KYT2016 project TURMET. It is complemented by the other deliverable of the project in 2015 written by Edoardo Tosoni titled “Scenario Analysis for the Safety Assessment of Nuclear Waste Repositories”.

In this report we take a general look at what a Safety Case is, what concepts does it relate to and a brief look at previously made safety cases in the field. The report also contains a section practical implications of safety cases as experienced in experimental work at VTT.
Some Safety Case terminology as used in nuclear waste management is listed in Table 1-1.

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1. Introduction

1.1 TURMET - Systemizing Safety Case Methodology

This literature review is made as a part of the KYT2018 project TURMET - Systemizing Safety Case Methodology. The topic is Safety Case in nuclear waste, or spent fuel, deposition.

There are many types of radioactive waste generated in medicine and other areas but the most hazardous one is spent nuclear fuel. Exposure to it will lead to death and it will remain radioactive for hundreds of thousands of years. In order to protect humans and the biosphere for its harmful effects it has to be isolated with extreme care for a long time to come, and the safety of the repository it is deposited in has to be proven by collecting as much scientific evidence as practicable in order to build a Safety Case that holds all the evidence and arguments on the subject.

The aim of the literature review is to take a comprehensive look at what is a Safety Case in general and in nuclear waste management in special, and how it has been used both in Finland and internationally. Although several methods for nuclear waste deposition have been suggested, deep geological repositories are the most universally accepted solution and the only one we will focus on in this report. The safety for a nuclear waste repository should be to demonstrated during all phases of its life cycle (construction, operation and closure) but the focus in this review is the post closure long term safety of the repository, which is also the most challenging aspect.

In the following Chapters we will first take a look at some general concepts regarding safety and the Safety Case and then move on to discussing the use of Safety Case in nuclear waste management. Some key countries and their methods are looked at as examples and the structure and methodology regarding Safety Case is reviewed, as well as some practical limitations and imperfections to be found in existing Safety Cases. It should be noted that Scenarios and Scenario development are not discussed in this review to any great length since this is done in the second TURMET deliverable by Edoardo Tosoni.
1.2 The Language of Safety

Before discussing the Safety Case itself we will take a brief look at the relevant terminology.

Firstly, words such as risk, chance and probability are often used interchangeably but actually have different meanings. (Maguire 2008) writes this in the following way:

“Consider the following three statements:

There is a risk of an accident. There is a chance of an accident. There is a probability of an accident. As the reader of these statements, ask yourself if there any (real) difference in these three statements, from your current understanding of what risk, chance and probability mean. One difference might be the potential numerical relationship between these non-numerate terms. Risk has the idea of something that might really happen, chance seems to imply that the something might not happen and probability certainly has the message that something will definitely happen.”

Thus the choice of words can have a significant impact on the reader and this should be kept in mind.

Some Safety Case terminology as used in nuclear waste management is listed in Table 1-1.

Table 1-1. Important Safety Case terminology (OECD NEA 2013)

<table>
<thead>
<tr>
<th>Safety Case</th>
<th>A formal compilation of evidence, analyses and arguments that quantify and substantiate a claim that the repository will be safe</th>
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<tr>
<td>Safety Context</td>
<td>A clear statement of purpose to set the safety case in its decision context including describing the decision being supported.</td>
</tr>
<tr>
<td>Safety Strategy</td>
<td>The high-level approach adopted for achieving safe disposal, including an overall management strategy, a siting and design strategy and an assessment strategy.</td>
</tr>
<tr>
<td>Assessment Basis</td>
<td>The collection of information and analysis tools supporting the safety assessment.</td>
</tr>
<tr>
<td>Safety Assessment</td>
<td>A methodological approach to the numerous processes, features and the chosen indicators such as dose or risk, using either mathematical analyses or more qualitative arguments.</td>
</tr>
<tr>
<td>Statement of Confidence</td>
<td>A statement by the authors of the Safety Case. Recognises any open issues and uncertainties.</td>
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2. Safety Case

2.1 The Definition of Safety Case

A safety case is a structured argument used in safety engineering to justify that a specific system is acceptably safe in its operating environment. This usually translates to an attempt at proving that the level of risk involved in operating the system is below certain limits for example given by a regulation authority. Safety cases are often required as a part of regulatory processes such in many fields, including nuclear waste management where the safety case plays an important role in the licensing processes but also gaining public acceptance. One definition for safety case is

“Safety Case: A structured argument, supported by a body of evidence that provides a compelling, comprehensible and valid case that a system is safe for a given application in a given operating environment.” (Maguire 2008)

It should be noted that the term “Safety Case” is to a degree of UK origin and for example in the US other terms such as “Safety Analysis Report” or “Safety Evaluation Report” may be more common.

2.2 Requirements for a Safety Case

A Safety Case should be:

- Transparent – clear and understandable for the intended audience
- Traceable – containing all of the necessary assumptions and data, either in itself or in supporting documents
- Open – any remaining uncertainties or open technical questions must be discussed
- Peer reviewed

2.3 The Purpose of a Safety Case

A Safety Case can serve many purposes. In general it is usually in answer to legal requirements; in many case (international) legislation and standards that state that obtaining a license to e.g. construct or operate a nuclear power plant requires a Safety Case to be built. However, primarily the purpose of a Safety Case is naturally to improve safety.

Safety Case can also be seen as

- Tracking the unavoidable risks involved. All risks can never be completely avoided (or the product or process would be completely safe in all circumstances) but the remaining risks must be estimated and analysed, and a Safety Case is a good place for it. Most countries have legal limits for risk exposure and the Safety Case can be used to check if the risk levels are legally acceptable.
- A management tool. Especially during longer projects (and nuclear waste management spans decades if not centuries) changes happen; the people, equipment and the technology change over time and original plans must be altered. It is essential that any change to the system does not lower its safety and the safety case for a long project should be constructed in such a way that those things that are likely to change can be evaluated later against their replacements.
A record of engineering practise. If something should go wrong the problem must be traced back to the decision that caused it and the Safety Case can be a tool for this. In addition to this, the Safety Case can also be useful if there is ever a reason to go to court, however, this is not the first concern when dealing with nuclear waste.

Marketing / PR tool. In some cases a product can be marketed through its rigorous approach to safety and Safety Case can play a big role in this. For NWM it is extremely important to portray to the general public that safety is taken very seriously and thus the Safety Case does not aim to only satisfy the legal requirements but also assure the populace that their safety is ensured to a very high degree.

In general a Safety Case can include a very large scope. What exactly is necessary is something that has to be decided at the start of constructing one based on which of these goals are necessary to be met. The purpose of the Safety case should be clear from the start so that they can act as the defining scope and boundary for it for a clear and logical end result.

2.3.1 Safety Boundary

Directly following from the scope and purpose of the Safety Case is a concept known as safety boundary. It defines all the processes, products and people that are involved in the safety analyses as well as assumptions made about the system and its environments. The boundary of things that need protection from harm extends from certain groups of people to local buildings and populace and biosphere as well as nation- or even worldwide consequences. In NWM this is often also referred to through timelines and different safety requirements may hold for the first 10 000 years vs. million years. This analysis requires identifying the interfaces through which the system can interact with its environment as well as identifying the essential scenarios in which the interactions can occur.

In constructing the safety boundaries it is very important that any limitations of the safety assessment are clearly stated e.g. in case of lacking or unclear data, limited experience or unknown operating conditions.

2.4 Safety Case in Nuclear Waste Management

2.4.1 Safety Aims of Nuclear Waste Deposition

The aims of nuclear waste disposal are as listed by IAEA (IAEA 2011):

- To contain the waste

- To isolate the waste from the accessible biosphere and to reduce substantially the likelihood of, and all possible consequences of, inadvertent human intrusion into the waste

- To inhibit, reduce and delay the migration of radionuclides at any time from the waste to the accessible biosphere

- To ensure that the amounts of radionuclides reaching the accessible biosphere due to any migration from the disposal facility are such that possible radiological consequences are acceptably low at all times.

These are the general aims for nuclear waste repositories. However, disposal facilities are not expected to provide full containment and isolation of waste over all time as this is neither practical nor required since the radiotoxicity of the waste is reduced over time. Thus more specific requirements are needed.
as a basis for legislation and consequently design. The international standard is set by IAEA (IAEA 2011) in their criteria for safety:

- The dose limit for members of the public for doses from all planned exposure situations is an effective dose of 1 mSv in a year. This and its risk equivalent are considered criteria that are not to be exceeded in the future.

- To comply with this dose limit, a disposal facility (considered as a single source) is so designed that the calculated dose or risk to the representative person who might be exposed in the future as a result of possible natural processes affecting the disposal facility does not exceed a dose constraint of 0.3 mSv in a year or a risk constraint of the order of 10−5 per year.

- In relation to the effects of inadvertent human intrusion after closure, if such intrusion is expected to lead to an annual dose of less than 1 mSv to those living around the site, then efforts to reduce the probability of intrusion or to limit its consequences are not warranted.

- If human intrusion were expected to lead to a possible annual dose of more than 20 mSv (see Ref. [7], Table 8) to those living around the site, then alternative options for waste disposal are to be considered, for example, disposal of the waste below the surface, or separation of the radionuclide content giving rise to the higher dose.

- If annual doses in the range 1–20 mSv (see Ref. [7], Table 8) are indicated, then reasonable efforts are warranted at the stage of development of the facility to reduce the probability of intrusion or to limit its consequences by means of optimization of the facility’s design.

- Similar considerations apply where the relevant thresholds for deterministic effects in organs may be exceeded.

These criteria are accompanied by the note that “radiation doses to people in the future can only be estimated and that uncertainties associated with these estimates will increase for periods farther into the future. Caution needs to be exercised in applying criteria for periods far into the future. Beyond such timescales, the uncertainties associated with dose estimates become so large that the criteria might no longer serve as a reasonable basis for decision making.” (IAEA 2011). This is typically interpreted so that at dose calculations are not done for very long time periods but instead the release rates of the nuclides are reviewed directly.

Many countries have also set stricter limits for e.g. dose limits.

The IAEA standard also includes non-radioactive safety considerations for e.g. environmental pollution or even disturbances during construction such as noise or traffic. However, this report focuses on the long term safety aspect of the Safety Case.

2.4.2 IAEA Safety Requirements for Nuclear Waste Deposition

The safety requirements are summarised in the following manner (IAEA 2011) (for the full description, the reader is advised to read the original document):

- **Requirement 1: Government responsibilities**
  Defining the national policy for the long term management of radioactive waste of different types. Establishing and maintaining an appropriate legal and regulatory framework for safety within which responsibilities shall be clearly allocated for disposal.

- **Requirement 2: Responsibilities of the regulatory body**
  Establishing regulatory requirements for the development of different types of disposal facility for radioactive waste and setting out the procedures for meeting the requirements for the various stages of the licensing process. It shall also set conditions for the development, operation and closure of each individual disposal facility.
• **Requirement 3: Responsibilities of the operator**
The operator is the responsible party for the safety of the repository. The operator must carry out safety assessment and develop and maintain the Safety Case, and carry out all the necessary activities for site selection and evaluation, design, construction, operation, closure and, if necessary, surveillance after closure.

• **Requirement 4: Importance of safety in the process of development and operation of a disposal facility**
An understanding of the relevance and the implications for safety of the available options for the facility shall be developed by the operator throughout the process of development and operation.

• **Requirement 5: Passive means for the safety of the disposal facility**
The operator shall evaluate the site and shall design, construct, operate and close the disposal facility in such a way that safety is ensured by passive means to the fullest extent possible and the need for actions to be taken after closure of the facility is minimized.

• **Requirement 6: Understanding of a disposal facility and confidence in safety**
The operator of a disposal facility shall develop an adequate understanding of the features of the facility and its host environment and of the factors that influence its safety after closure over suitably long time periods, so that a sufficient level of confidence in safety can be achieved.

• **Requirement 7: Multiple safety functions**
The host environment shall be selected, the engineered barriers of the disposal facility shall be designed and the facility shall be operated to ensure that safety is provided by means of multiple safety functions. Containment and isolation of the waste shall be provided by means of a number of physical barriers of the disposal system. The performance of these physical barriers shall be achieved by means of diverse physical and chemical processes together with various operational controls. The capability of the individual barriers and controls together with that of the overall disposal system to perform as assumed in the safety case shall be demonstrated. The overall performance of the disposal system shall not be unduly dependent on a single safety function.

• **Requirement 8: Containment of radioactive waste**
The engineered barriers, including the waste form and packaging, shall be designed, and the host environment shall be selected, so as to provide containment of the radionuclides associated with the waste. Containment shall be provided until radioactive decay has significantly reduced the hazard posed by the waste. In addition, in the case of heat generating waste, containment shall be provided while the waste is still producing heat energy in amounts that could adversely affect the performance of the disposal system.

• **Requirement 9: Isolation of radioactive waste**
The disposal facility shall be sited, designed and operated to provide features that are aimed at isolation of the radioactive waste from people and from the accessible biosphere. The features shall aim to provide isolation for several hundreds of years for short lived waste and at least several thousand years for intermediate and high level waste. In so doing, consideration shall be given to both the natural evolution of the disposal system and events causing disturbance of the facility.

• **Requirement 10: Surveillance and control of passive safety features**
An appropriate level of surveillance and control shall be applied to protect and preserve the passive safety features, to the extent that this is necessary, so that they can fulfil the functions that they are assigned in the safety case for safety after closure.

• **Requirement 11: Step by step development and evaluation of disposal facilities**
Disposal facilities for radioactive waste shall be developed, operated and closed in a series of...
steps. Each of these steps shall be supported, as necessary, by iterative evaluations of the site, of the options for design, construction, operation and management, and of the performance and safety of the disposal system.

- Requirement 12: Preparation, approval and use of the safety case and safety assessment for a disposal facility
A safety case and supporting safety assessment shall be prepared and updated by the operator, as necessary, at each step in the development of a disposal facility, in operation and after closure. The safety case and supporting safety assessment shall be submitted to the regulatory body for approval. The safety case and supporting safety assessment shall be sufficiently detailed and comprehensive to provide the necessary technical input for informing the regulatory body and for informing the decisions necessary at each step.

- Requirement 13: Scope of the safety case and safety assessment
The safety case for a disposal facility shall describe all safety relevant aspects of the site, the design of the facility and the managerial control measures and regulatory controls. The safety case and supporting safety assessment shall demonstrate the level of protection of people and the environment provided and shall provide assurance to the regulatory body and other interested parties that safety requirements will be met.

- Requirement 14: Documentation of the safety case and safety assessment
The safety case and supporting safety assessment for a disposal facility shall be documented to a level of detail and quality sufficient to inform and support the decision to be made at each step and to allow for independent review of the safety case and supporting safety assessment.

- Requirement 15: Site characterization for a disposal facility
The site for a disposal facility shall be characterized at a level of detail sufficient to support a general understanding of both the characteristics of the site and how the site will evolve over time. This shall include its present condition, its probable natural evolution and possible natural events, and also human plans and actions in the vicinity that may affect the safety of the facility over the period of interest. It shall also include a specific understanding of the impact on safety of features, events and processes associated with the site and the facility.

- Requirement 16: Design of a disposal facility
The disposal facility and its engineered barriers shall be designed to contain the waste with its associated hazard, to be physically and chemically compatible with the host geological formation and/or surface environment, and to provide safety features after closure that complement those features afforded by the host environment. The facility and its engineered barriers shall be designed to provide safety during the operational period.

- Requirement 17: Construction of a disposal facility
The disposal facility shall be constructed in accordance with the design as described in the approved safety case and supporting safety assessment. It shall be constructed in such a way as to preserve the safety functions of the host environment that have been shown by the safety case to be important for safety after closure. Construction activities shall be carried out in such a way as to ensure safety during the operational period.

- Requirement 18: Operation of a disposal facility
The disposal facility shall be operated in accordance with the conditions of the licence and the relevant regulatory requirements so as to maintain safety during the operational period and in such a manner as to preserve the safety functions assumed in the safety case that are important to safety after closure.

- Requirement 19: Closure of a disposal facility
A disposal facility shall be closed in a way that provides for those safety functions that have been shown by the safety case to be important after closure. Plans for closure, including the
transition from active management of the facility, shall be well defined and practicable, so that
closure can be carried out safely at an appropriate time.

- **Requirement 20: Waste acceptance in a disposal facility**
  Waste packages and unpackaged waste accepted for emplacement in a disposal facility shall
  conform to criteria that are fully consistent with, and are derived from, the safety case for the
  disposal facility in operation and after closure.

- **Requirement 21: Monitoring programmes at a disposal facility**
  A programme of monitoring shall be carried out prior to, and during, the construction and
  operation of a disposal facility and after its closure, if this is part of the safety case. This
  programme shall be designed to collect and update information necessary for the purposes of
  protection and safety. Information shall be obtained to confirm the conditions necessary for the
  safety of workers and members of the public and protection of the environment during the period
  of operation of the facility. Monitoring shall also be carried out to confirm the absence of any
  conditions that could affect the safety of the facility after closure.

- **Requirement 22: The period after closure and institutional controls**
  Plans shall be prepared for the period after closure to address institutional control and the
  arrangements for maintaining the availability of information on the disposal facility. These plans
  shall be consistent with passive safety features and shall form part of the safety case on which
  authorization to close the facility is granted.

- **Requirement 23: Consideration of the State system of accounting for, and control of,
  nuclear material**
  In the design and operation of disposal facilities subject to agreements on accounting for, and
  control of, nuclear material, consideration shall be given to ensuring that safety is not
  compromised by the measures required under the system of accounting for, and control of,
  nuclear material.

- **Requirement 24: Requirements in respect of nuclear security measures**
  Measures shall be implemented to ensure an integrated approach to safety measures and
  nuclear security measures in the disposal of radioactive waste.

- **Requirement 25: Management systems**
  Management systems to provide for the assurance of quality shall be applied to all safety
  related activities, systems and components throughout all the steps of the development and
  operation of a disposal facility. The level of assurance for each element shall be commensurate
  with its importance to safety.

- **Requirement 26: Existing disposal facilities**
  The safety of existing disposal facilities shall be assessed periodically until termination of the
  licence. During this period, the safety shall also be assessed when a safety significant
  modification is planned or in the event of changes with regard to the conditions of the
  authorization. In the event that any requirements set down in this Safety Requirements
  publication are not met, measures shall be put in place to upgrade the safety of the facility,
  economic and social factors being taken into account.

2.4.3 Safety Case and Repository Lifecycle

Building a Safety Case for nuclear waste management differs from many other fields in that the
timescales are enormous and the possible future conditions of the repository difficult to predict with
accuracy. Demonstrating the long term safety performance of the repository with confidence under
these circumstances is not easy as the possible evolutions of the geological site, host rock and the engineered barrier system all have to be investigated.

In The Nature and Purpose of the Post-closure Safety Cases for Geological Repositories (OECD NEA 2013) it is stated that

“A safety case is a formal compilation of evidence, analyses and arguments that quantify and substantiate a claim that the repository will be safe. An initial safety case can be established early in the course of a repository project. Such a preliminary safety case then evolves into a more comprehensive safety case as a result of work carried out, incorporating experience gained and information obtained throughout the stepwise evolution of the project.”

Thus in nuclear waste management safety case building is recognised as an iterative process that evolves during repository design and development. Early safety cases may be built on generic assumptions on e.g. host rock properties (for example, see Chapter 3.3 for the Japanese Safety Case) used as help in site characterisation and selection process whereas the Safety Case used for licensing purposes has to be much more detailed and rigorous. After construction the Safety Case should be updated with any new information learned during the construction. The lifecycle phases of a nuclear waste repository are summarised below. It should be noted that the names for the categories and the classification varies between sources.

- **(National framework preparation)**
  Setting up the legislation, regulations and organisations to enable licensing of nuclear waste management systems.

- **Pre-construction phase**
  The waste disposal concept is selected and siting decisions are made. This phase involves a lot of research into conditions on the site and the long term safety of the final repository concept. The Safety Case will likely go through some iterations within this phase as after the siting decision the site specific properties may be properly included.

- **Construction phase**
  The repository design is specified, designed and implemented, during the creation stage. Any new information gained during the construction process and its implications must be included in the Safety Case.

- **Operational phase**
  Many of the processes mentioned above continue and the disposal of the nuclear waste begins. The repository operation will adopt the nuclear safety culture with the works encouraged to question and criticism, likely leading to changes and improvements in design and processes. The Safety Case will be updated accordingly.

- **Retirement stage**
  Closure of the repository after all of the waste has been deposited, and of the related operational and support processes.

- **Post-closure stage**
  After the repository is closed and sealed off it will be initially monitored and sealed off from general use. However, the safety of the repository cannot rely on active measures for the long term since the time span is so enormous.
As mentioned above, the Safety Case should be updated at regular intervals in all of these phases.

2.4.4 Safety Case Elements

The Safety Case in nuclear waste deposition is built on a number of elements that may vary from case to case as per national legislation. One illustration of the elements is given in Figure 2-1 as given by OECD (OECD NEA 2013).

*Figure 2-1. The structure of elements in a Safety Case (OECD NEA 2013).*
2.4.5 Long Term Safety

During its operation many safety aspects of the repository are important such as security, radiological and occupational safety of workers and protection of the public and environment from potential radiological exposures. However once the repository is closed and active control of the facility ceases the focus becomes on long term safety assessment. This means in practise analysing the long term containment of the hazardous nuclides and the circumstances in which they may be released from the repository, in which quantities, how likely these releases are how they can be prevented, limited or mitigated and what the consequences of the releases are to humans or the biosphere.

3. Examples of Safety Cases in Nuclear Waste Management

3.1 Finland

The Posiva Safety Case is discussed in Chapter 4.

3.2 Switzerland

A total of 3.6 tonnes of spent fuel is expected from the current Swiss reactors during their lifetime. The Swiss government passed a law in 2006 setting the roles for nuclear waste deposition, the responsibility is on the licence holders. The fate of the spent fuel is currently not decided, fuel reprocessing has been on a 10 year moratorium since 2006 due to legislation. The Swiss implementer of the deposition is Nagra.

The Swiss law requires a demonstration of disposal feasibility. This was implemented and approved by the federal government in June 2006. The demonstration was based on a deep geological repository concept with tunnel emplacement of spent fuel and high level waste as well as caverns for long lived intermediate level waste. Early siting studies focused on the crystalline basement of Northern Switzerland. The authorities also required investigation of sedimentary formations. After an iterative process, the Opalinus Clay and Zürcher Weinland in the northern part of the Swiss Plateau as a model siting region was selected and a detailed characterisation programme was carried out including seismic studies, exploratory borehole research and extensive studies on Opalinus Clay. Based on these results Nagra submitted the demonstration of disposal feasibility and it was approved in June 2006. In 2008 they submitted three alternative sites for the repository of spent fuel, and the siting process is still ongoing. Deposition is expected to start in 2035 at earliest. (OECD NEA 2014)

Nagra’s safety case considers the risk on the critical group of people during the next million years. There are some specific requirements such as controllability (post operational control and in situ monitoring of the repository) and retrievability. Nagra defines safety criteria (isolation, long term confinement, attenuation of releases) and “pillars of safety” (deep underground location, host rock properties, chemical environment, bentonite buffer properties, stable fuel matrix and mechanically strong canister) that provide the safety functions. They also divide their safety principles into three categories; for the overall system, for the specific site and for the specific repository design. (Junker et al. 2008)

3.3 Japan

The Japanese have published a conference article on their Safety Case development during 2015 (Fujiyama et al. 2015) addressing the key challenges and advances on the subject and this will be briefly summarized here.

In 2011 Japan was hit by the disastrous earthquake and tsunami leading into one of the worst nuclear accidents ever in Fukushima. As a result, nuclear safety has been a very critically viewed topic in Japan
and there are serious concerns over the safety of deep geological disposal due to the seismic and volcanic activity of the Japanese bedrock. Multidisciplinary Working Groups were established to review geological disposal and its implementation process. The conclusions were that

“1) Potentially favourable geological environments for geological disposal exist and selecting them is feasible, even considering the latest geoscientific knowledge

2) Periodic re-evaluation of safety based on the latest knowledge and communicating this with the general public are inevitable.” (Fujiyama et al. 2015)

Following this conclusion the Government decided in 2014 to promote site selection based on scientific knowledge of suitable sites, which may solve the deadlock site selection has been in in Japan. Consequently the Nuclear Waste Management Organization of Japan (NUMO) is currently developing a Safety Case (referred to as the NUMO 2015 Safety Case) in accordance with OECD/NEA and IAEA guidelines. Due to site selection not having been done the Safety Case uses generic synthesized geological and hydrogeological models of host rock based on gathering information on the potential sites. The repository concept is built on the usual engineered barrier system with multiple barriers in a mined geological repository, and the repository will contain 40,000 canisters of vitrified HLW.

The NUMO 2015 Safety Case describes up a safety strategy as an implementation policy bearing in mind that some safety regulations are still under development. The Safety Case sets up a management strategy, siting and design strategy and safety assessment strategy.

The management strategy handles topic such as coordination between different topics and knowledge transfer during the project:

- Coordination and integration among different technology fields
- Iterative confirmation of safety and development of a safety case
- Risk management
- Quality management
- Knowledge management
- Management of the R&D program
- Human resources management strategy
- Strategy on communication with stakeholders.

The siting strategy describes the process of site selection as literature survey, preliminary investigation and detailed investigation. Explicit exclusion criteria following from geological attributes based on tectonic activity are included. Illustrative site descriptive models will be developed for repository design and safety assessment.

The design strategy describes facilities based on the site descriptive models that fulfil the expected safety requirements. The design will be improved step by step alongside of the site selection process and this section will also include verification and vilification of the design technologies. The NUMO 2015 Safety Case will present alternative design methodologies (alternative repository concepts) to maintain flexibility to adapt for a wide range of geological conditions.

The safety assessment strategy involves both the pre- and post-closure safety of the repository. Similarly to the design strategy, the safety assessment strategy will be developed in an iterative manner alongside with the site selection process. Regulations for operational safety in a nuclear repository do
not exist yet so the regulations for other nuclear facilities as well as mining and civil engineering safety regulations will be used as a guideline. For post-closure safety a lot of effort has been used on scenario selection, and this will be discussed in the TURMET deliverable on Scenario Analysis.

4. Practical Experiences of Safety Case Limitations and Uncertainties

A survey was conducted based on internal interviews at VTT in group BA2112 and published reports by Posiva with the aim at investigating how the Safety Case connects to design and manufacturing. The interviewed researchers work mainly in EBS related studies, such as buffer and backfill design, plug design, manufacturing of bentonite blocks and pellets, instrumentation techniques and small, middle and full scale demonstrations of EBS. Also site selection and characterization and geochemical modelling belong to interest areas in group BA2112.

The interviewed researchers and their personal expertise were:

- Edgar Bohner (Buffer block manufacturing, instrumentation and demonstration of the plug)
- Kai Front (Site selection and characterization)
- Erika Holt (Plug design, plug demonstration in full scale, buffer block and pellet manufacturing, small scale demonstrations)
- Markku Juvankoski (Buffer design, backfill design)
- Harri Kivikoski (Middle and full scale demonstrations of EBS, pellet manufacturing)
- Jutta Peura (Backfill block manufacturing, moisture protection system design, mock-up test for EBS)

This survey includes practical examples of design, manufacturing and demonstration of EBS in KBS-3V concept that are connected to safety case and scenario analyses for a final repository expect the copper canister and host rock.

The results of interview are based on researcher’s own experience, opinions and assumptions but they may not represent views of whole nuclear waste management industry. It must be also noted that researchers who were involved into this study work mainly in projects that are related to manufacturing, implementation or/and demonstrations of EBS so they are following the long-term safety requirements or scenario analyses only their perspective.

4.1 Engineer Barrier System (EBS)

In deep geological disposal systems, the barriers include the natural geological barrier and the Engineer Barrier System (EBS). The EBS contains a variety of sub-system or components, such as canister, buffer, backfill and closure. The objective of EBS is to prevent and/or delay the release of radionuclides from the waste to the repository host rock (OECD, 2005). The host rock should provide favourable conditions for the long-term performance of the engineered barriers, but also limit or retard the transport of radionuclides. The multi-barrier system as a whole should be able to protect the living environment even if one of the barriers turns out to be deficient (Posiva 2012-12).

In Posiva’s KBS-3V concept the canister is the first barrier, buffer is second barrier, and backfill is third barrier and so on. For each component have its own requirements to fulfil. The safety functions of EBS and host rock in Posiva’s KBS-3V concept are given in Table 2.

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<td>Canister</td>
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| Buffer    | • Contribute to mechanical, geochemical and hydrogeological conditions that are predictable and favourable to the canister.  
• Protect canisters from external processes that could compromise the safety function of complete containment of the spent nuclear fuel and associated radionuclides  
• Limit and retard radionuclide releases in the event of canister failure. |
| Backfill  | • Contribute to favourable and predictable mechanical, geochemical and hydrogeological conditions for the buffer and canisters.  
• Limit and retard radionuclide releases in the possible event of canister failure.  
• Contribute to the mechanical stability of the rock adjacent to the deposition tunnels. |
| Closure   | • Prevent the underground openings from compromising the long-term isolation of the repository from the surface environment and normal habitats for humans, plants and animals.  
• Contribute to favourable and predictable geochemical and hydrogeological conditions for the other engineered barriers by preventing the formation of significant water conductive flow paths through the openings.  
• Limit and retard inflow to and release of harmful substances from the repository. |
| Host rock | • Isolate the spent nuclear fuel repository from the surface environment and normal habitats for humans, plants and animals and limit the possibility of human intrusion, and isolate the repository from changing conditions at the ground surface.  
• Provide favourable and predictable mechanical, geochemical and hydrogeological conditions for the engineered barriers.  
• Limit the transport and retard the migration of harmful substances that could be released from the repository. |
4.2 Requirement management

There are huge amounts of different levels of requirements for each component of EBS. Posiva has developed its own requirement management system called VAHA which is an information system designed in Posiva to manage the requirements related to the geological disposal of spent nuclear fuel. VAHA aims to include all relevant requirements, origin and their rationale with existing solutions to fulfill them, and enables an effective review of compliance and dependencies between separate specifications and requirements. The VAHA database is organised into five levels (Posiva 2012-12):

I. Level 1 consists of the Stakeholder requirements. These are the requirements arising from laws, regulatory requirements, decisions-in-principle and other stakeholder requirements.

II. Level 2 consists of the System requirements as defined by Posiva on the basis of Posiva's owners' requirements and the legal and regulatory requirements listed on Level 1. Level 2 requirements define the EBS components and the functions of the EBS and host rock.

III. Level 3 consists of the Subsystem requirements which are specific requirements for the canister, buffer, backfill, closure and host rock and underground openings. The requirements of level 3 are mostly general and set qualitative requirements (performance targets and target properties) for EBS and host rock performance.

IV. Level 4 Design requirements further clarify and provide more details to the requirements of Level 3.

V. Level 5 presents the Design specifications. These are the detailed specifications to be used in the design, construction and manufacturing.

Each requirement in the VAHA system has an identification number based on the requirement level, system, and requirement number. For example, L4-CAN-2 relates to the second Level 4 requirement on the canister. The Posiva design basis contains all requirements for Levels 1 to 4 of the VAHA system. Production Line Reports present the Level 5 requirements (design specifications) for all EBS components.

The VAHA system has been formulated between the decisions-in-principle for disposal of spent nuclear fuel and the repository construction licence application in response to a need for organizing the requirements. VAHA requirements have been formulated as a group effort in Posiva with the responsible persons for subsystem development having a large role. Changes to VAHA can be applied by making a “decision guiding requirements” (this is referred to in Finnish as Vaatimuksia ohjaava päätös (VOP)). A VOP application is made using a template that presents the current requirement, suggestions for changes and the rationale for the change. VOPs are presented to the Posiva Technical Group for acceptance. Design personnel take part in the updating of requirements and the change management procedure, and, therefore, requirements management is undertaken in parallel with design development.

The Production Line reports provide supporting information to the construction licence application and these are included in STUK’s on-going review. STUK’s review of the construction licence application is expected to include review of, and feedback on, the maturity of the repository subsystems. This may lead to further development of the design basis.
4.3 Buffer Design

The buffer is the second of the barriers of the KBS-3 method and the safety functions of it are presented in Table 2. According to Posiva 2012-03 the requirements can be divided into two categories: requirements related to the performance of the buffer, and requirements related to support to other systems. The performance requirement can be further divided into requirements related to chemical protection, mechanical protection and limitation of mass flows in the near field, and heat transfer, which is needed for EBS compatibility.

In VAHA level 1 the Buffer is defined as following: the component that surrounds the canister and fills the void spaces between the canister and the rock. The purpose of the buffer is to protect the canister from detrimental thermal, hydraulic, mechanical and chemical, including microbiological (THMC) processes that could compromise the safety function of complete containment, to maintain favourable conditions for the canister and to slow down the transport of radionuclides if the canister starts leaking. Requirements for the buffer in VAHA levels 3-4 are listed below:

**Performance (VAHA, level 3)**

- The amount of substances in the buffer that could adversely affect the canister, backfill or rock shall be limited.
- Unless otherwise stated, the buffer shall fulfil the requirements listed below over hundreds of thousands of years in the expected repository conditions except for incidental deviations.
- The buffer shall transfer the heat from the canister efficiently enough to keep the buffer temperature < 100°C.
- The buffer shall mitigate the impact of rock shear on the canister.
- The buffer shall allow gases to pass through it without damage to the repository system.
- The buffer shall limit microbial activity.
- The buffer shall mitigate the impact of rock shear on the canister.
- The buffer shall be impermeable enough to limit transport of radionuclides from the canister into the bedrock and the buffer shall be impermeable enough to limit the transport of corroding substances from the rock onto the canister surface and The buffer shall limit the transport of radio colloids.

**Support of the other system (VAHA, Level 3)**

- The buffer shall provide support to the deposition hole walls to mitigate potential effects of rock damage.
- The buffer shall be able to keep the canister in the correct position (to prevent sinking and tilting).

**Design requirements for the buffer (VAHA, Level 4)**

- The main component of the buffer material shall consist of natural swelling clays.
- The buffer shall be designed to be self-sealing after initial installation and self-healing after any hydraulic and mechanical disturbances.
- The buffer shall be so designed that the possibility of corrosion of a canister by sulphide and other corrodants including microbially-induced processes will be limited and The buffer material shall be selected so as to limit the contents of harmful substances (organics, oxidising compounds, sulphur and nitrogen compounds) and microbial activity.
- The buffer shall be so designed that it will mitigate the mechanical impact of the postulated rock shear displacements on the canister to the level that the canister integrity is preserved.
- The buffer shall be designed in such a way as to make diffusion the dominant transport mechanism for solutes.

*and*
The buffer material must be selected in a way that favours the retardation of the transport of radionuclides by sorption (e.g. cation exchange) at the clay and other mineral surfaces.

The buffer shall have sufficiently fine pore structure so that transport of radio colloids formed within or around the canister is limited.

- The gap between the canister and buffer and buffer blocks and rock should be made as narrow as possible without compromising the future performance of the buffer.
- The buffer shall initially provide a good contact with the host rock.

In the actual buffer design all mentioned requirements and safety functions are taken account and Posiva’s reference design for the buffer is described in working report called Description of Basic Design for Buffer by Juvankoski (2009) and Buffer Design Basis by Juvankoski and Marcos (2009).

The Buffer Design Basis covers specifications for buffer design, the role of buffer, site-specific data concerning the buffer, requirements and results from previous design, for example fixed parts of the system, which shall be taken into account in the buffer design, in the interfaces of the buffer with other barriers (backfill, rock and canister), in manufacturing, and in the buffer emplacement in the deposition holes. An important starting point for the Buffer Design Basis has been the regulatory framework and the design requirements related to long-term safety. The Buffer Design Basis also takes into account the evolving conditions and expected loadings on the buffer in the repository after emplacement to ensure the appropriate long-term performance of the buffer and thus long-term safety (Juvankoski & Marcos, 2009).

According to Juvankoski (2009) single canisters containing spent nuclear fuel are emplaced in individual vertical deposition holes drilled in the floor of deposition tunnels in the bedrock at about 420 m depth in the KBS-3V repository concept. The diameter of the deposition holes are 1.75 m and the depths from 6.60 m to 8.25 m. In the basic design disk type bentonite blocks are installed at the bottom of the hole and on top of the spent fuel canister. Ring type bentonite blocks surround the fuel canister. In the basic design MX-80 bentonite is the reference bentonite material. The gap between the deposition hole rock surface and the bentonite blocks is 25 mm and the gap between the bentonite blocks and fuel canister is 10 mm. In the basic design the gaps will not be filled at emplacement time.

Bentonite blocks have a diameter of 1.70 m and a height of 0.40 m. At installation the water content of blocks is about 16% and the bulk density of blocks is about 2100 kg/m$^3$. At target state the saturated density in the buffer is about 2000 kg/m$^3$. In the basic design the swelling of bentonite is assumed to be achieved due to inflowing water leaking from the rock over time. (Juvankoski, 2009).

If the amount of inflowing water from the rock is small, the swelling of bentonite can be delayed. The increase of temperature of the rock could result in the spalling of the upper parts of the deposition hole, if the buffer does not exert swelling pressure on it. On the other hand, if water inflow to the hole is high, it can cause problems with the installation and may cause erosion of bentonite. It is important to be prepared for both dry and wet holes\(^1\). This in turn affects the installation schedule. (Juvankoski, 2009)

In order to ensure the successful installation and behaviour of the buffer the details of the deposition hole are important, e.g. wall surface flatness, evenness of the base and the chamfer required on the top of the hole for the disposal of spent fuel canisters of the Olkiluoto nuclear power plants. All of these must be designed so that the requirements set for the buffer are met. (Juvankoski, 2009)

So the buffer design is answering the questions like:

- Used material type and its chemical composition
- Geometry of buffer components (such as blocks, pellets, gaps)
- Geometry of buffer and interfaces (host rock, backfill)

\(^1\) The maximum inflow is 0.1 l/min in deposition holes. If the flow is higher the deposition hole will be rejected.
- Target saturated density value of the used material
- Evaluation of the initial state of the buffer
- Manufacturing, installation and quality assurance aspects

The design of the buffer is an iterative process which is developing based on latest studies and gathered knowledge. For example the design of the moisture protection system (Ritola & Peura, 2012) and isostatic buffer block manufacturing experiments (Ritola & Pyy, 2011) have changed the reference buffer design (Juvankoski, 2009; Juvankoski & Marcos, 2009) as described in Buffer Design (Juvankoski, 2012):

- The reference design now also contains the temporary moisture protection system for the buffer during the installation phase.\(^2\)
- The reference design for the bottom of the deposition hole is now an evened rock surface with a copper plate covering instead of levelling the bottom with a low pH mortar slab.\(^3\)
- The outer gap width between the bentonite blocks and rock wall is increased from 25 mm to 50 mm.\(^4\)
- The diameter of bentonite blocks is reduced from 1700 mm to 1650 mm to match the increased gap.\(^5\)
- The outer gap between the bentonite blocks and rock wall is filled with bentonite pellets.\(^6\)
- The heights of the bentonite blocks around the canister are changed from standard measures to canister specific measures.\(^9\)
- The height of the bentonite block below the canister is changed from 800 mm to 500 mm.\(^7\)
- The height of the bentonite block pile above the canister is changed from 2200 mm to 2500 mm.\(^8\)
- In the depth of the deposition hole the tolerance of the tunnel floor and the thickness of the copper plate are taken into account.\(^9\)
- The water content of the buffer blocks and pellets are changed, and are now all 17\%.\(^7\)
- The bulk densities of the blocks are changed to be block type specific.\(^9\)
- The chamfer shape for OL1-2 and OL3 deposition holes are changed to be crescent shaped profiles rather than wedge shaped.\(^8\)
- The montmorillonite content of buffer material has a set upper limit of 90\%.\(^9\)

The long-term function of the buffer is based on the material properties and predictable behaviour of selected material. The most important factors effecting to the target density are hydraulic conductivity, load-bearing capacity and swelling pressure. In reference design (2012) an adequate swelling pressure is lower than 15 MPa (in earlier design it was 2-10 MPa). According to Juvankoski (2012) external mechanical loads to buffer come from the natural environment and from the weight of the canister, the buffer itself and the tunnel backfilling. The nominal depth of the Olkiluoto reference repository is -420 m. Thus the maximum hydrostatic groundwater pressure is 4.1 MPa. The maximum postulated ice layer...
during glaciation at Olkiluoto area is 2.5 km thick, according to Pimenoff et al. (2011) and Lambeck & Purcell (2003). This 2 km ice sheet may create a pressure of about 25 MPa to the groundwater pressure, if the effect of the ice layer is conservatively added to the hydrostatic pressure. These isostatic loads do not affect harmfully to the buffer. Freezing and thawing are not expected to affect the buffer or backfill as climatic conditions leading to permafrost at repository depth are not expected (Hartikainen 2012).

Nonetheless this process is taken into account in the selection of the materials used; the performance of the buffer and backfill materials under freezing-thawing conditions has been reported by Schatz & Martikainen (2010) with the conclusion that freeze-thaw cycles do not substancially degrade the buffer and backfill material properties.

In the buffer design all requirements and safety functions as well as technical, geometrical, manufactural, installing, timing and excavation aspects are formed as numerical values of the buffer. As shown in given example results of other studies can cause a lot of changes in actual design. It is important that all results affecting to EBS are delivered between interest group as soon as possible so that any aspect do not pass unnoticed.

In fact, every component of EBS has develop and changed in course of time as the similar way than the buffer design but changes have not listed as clearly. At this point, design and develop of EBS components are still on-going for the final disposal and will continue even if the repository is already started. So it can be said that the design is continuous change management.

4.4 Manufacturing, especially backfill blocks

Requirements related to the manufacturing of buffer or backfill blocks are come from the design, production and initial state of the buffer or backfill. The most significant parameters from the manufacturing perspective are nominated material, material properties such as grain size distribution, target density and used water content values and of course the dimensions of the manufactured block. For example Posiva’s design parameters for backfill and buffer blocks are listed below:

**Buffer block**

Material: MX-80 type of bentonite  
Dimensions:  
- outer diameter: 1 650 mm (cylindrical and ring blocks)  
- inner diameter: 1 070 mm (ring blocks)  
- height: 500-875 mm  
Target density for fully saturated buffer: 2 000 ± 50 kg/m$^3$  
Initial water content: 17 %  
Manufacturing tolerances: no specified because buffer blocks will be machined after manufacturing  
Manufacturing method: isostatic or uniaxial (decision is still pending)

**Backfill block**

Material: Friedland clay  
Dimensions: 330 x 470 x 550 mm  
Target dry density: 2 030 ± 40 kg/m$^3$  
Initial water content: 9 ± 0.5 %  
Manufacturing tolerances: -1 to +2 mm (no machining)  
Manufacturing method: uniaxial

In addition to these numerical values different manufacturing methods and their possibilities and limitations, such as production capacity and rate, adjustment ranges of pressing machine and post-
processing methods\textsuperscript{10}, is needed to know well. It is also crucial to understand the behaviour of material before, during and after the compaction process, for example the segregation of raw bentonite material during its transportation and handling (before the compaction), air escaping during the compaction\textsuperscript{11} and the relaxation\textsuperscript{12} of the block after compression. So the material quality control and assurance have an important role in manufacturing process (Holt et al., 2014; Peura et al., 2015).

The goal of manufacturing process despite of chosen manufacturing method is to procure high quality bentonite blocks which means that all given numerical parameters are achieved and the surface and inside of the block are undamaged (no visible cracks). To produce bentonite blocks without cracks would be much easier and more cost-effective if lubricants or different kinds of coatings are allowed but it is possible that lubricants get trapped within the bentonite during the mould filling and thus may contaminate the deposition hole. The compressed bentonite surfaces may have to machine away which is money and time consuming. Permanent coatings are a more viable option, though their abrasive resistance should be further verified or tailored by the manufacturers before full-scale production. (Holt & Peura, 2011). Posiva has decided that any lubricants or coatings are not allowed in full scale production of bentonite blocks so that the long-term safety do not jeopardize. The full scale production tests have shown that it is possible to produce high quality bentonite blocks\textsuperscript{13} by both methods but cracks are still occurred in the surfaces of blocks and the production rate of the blocks is not as high as desired (Peura et al., 2015).

The production of the bentonite blocks is controlled by many predetermined parameters, some of which are connected straight to design and some of which are related to the technical requirements which do not have a visible link to the long-term safety. Sometimes requirements are set based on hypotheses of experts when the actual results are not available.

A similar example can be given also from changing the corners of the mould that is used in the backfill manufacturing. Initially, the corners of the mould were exact 90 degrees (square). Since then it was noticed that sharp corners of the manufactured blocks were too brittle and they were crumbled easily. So a new mould was designed with rounded\textsuperscript{14} corners. This had a positive effect to the quality of the backfill blocks because the corners were now more permanent and they were not broken so easily anymore (Holt et al., 2014; Peura et al., 2015).

As mentioned before, in the production of bentonite blocks must be taken into consideration also other aspects than only design requirements. For example, buffer blocks manufactured by isostatic compression method are needed to machine afterwards. This means that the production of blocks with this method contains one phase more than uniaxial method. Due to machining of the buffer blocks the material loss is typically 5-10 % and surplus material need to utilize (Ritola & Pyy, 2011). On the other hand, the quality of isostatic buffer blocks can be proven better than uniaxially produced blocks. Posiva has not decided yet by which method it will prepare the buffer blocks because studies related to this topic are still on-going.

Sometimes in the light a new scientific knowledge and studies, the radical changes must be done in design of the blocks.

In practical level the selection of new reference material means a lot new calculations and studies that are listed below (only main topics)\textsuperscript{15}:

\textsuperscript{10} For example protection, packing and storage of the blocks, but also material loss if blocks are machined. Surplus material should be utilized after compression, for example in pellet manufacturing. (Ritola & Pyy, 2011; Peura et al., 2015)

\textsuperscript{11} If the compaction is too fast air does not have time to escape and it will cause cracks and joints to the blocks. (Holt & Peura, 2011; Holt et al., 2014; Peura et al., 2015)

\textsuperscript{12} Dimensions of the manufactured block are grown after compaction in few minutes. It is due to the initial properties of bentonite material. (Holt & Peura, 2010; Holt et al., 2014; Peura et al., 2015)

\textsuperscript{13} Depending on bentonite material type (with MX-80 this is possible, Friedland clay not)

\textsuperscript{14} Rounding are 5 degrees in every corner (Holt et al., 2014)

\textsuperscript{15} The list is general enough to use for this purpose but explanations for changing of Posiva’s reference materials or even mentioning of it is highly confidential before the working reports are published.
- Chemical and physical properties of different material candidates are needed to study
- Mapping the possible supplier of materials and evaluate a dependability of supplier and the quality of delivered material in long-term customer relationship (supplier are around the world) and of course agree prices, degree of processing and shipping
- Quality control of raw materials (grain size distribution for instance) and the characterization of candidates
- Defining the new target dry density values for every material candidate
- Small and middle scale tests are needed to evaluate a right water content level so that the target density will fulfill the most likely in factory scale trials.
- Factory scale trials for each material
- Quality control of the manufactured blocks
- Evaluating the material candidates based on laboratory, middle and full-scale tests
- Evaluating the pellet manufacturing on based back fill studies16
- Changing the original design of pellets and/or making new studies with a reference material17
- Decision-making

So the whole procedure will take a lot of time and money.

It must be also noted that the planning of the actual manufacturing facilities have not even designed yet. On that perspective some limitations can occur but they are probably solved by technical solutions.

A summary/conclusion: technical requirements or manufacturing processes can change the original design of bentonite blocks if necessary but they do not change the requirements of the long-term safety.

4.5 Demonstrations

The purpose of demonstrations is to show that design and implementation plans of EBS are performed and behaved as expected and the long-term safety requirements are fulfilled. Demonstrations provide also valuable information about actual behaviour of EBS and unrecognized actions that may occur during the testing. Demonstrations are performed in different scales (small, medium and full), in different environments (laboratory, in-situ), and from different intentions and purposes. In this context few examples are given about different scales demonstrations that have been done at VTT.

**Artificial Wetting Tests (2009-13), small scale**

The canisters containing spent nuclear fuel are surrounded by blocks and rings of compacted bentonite. The function of this bentonite buffer is to restrict groundwater flow and to protect the canister. This bentonite engineering barrier system has to fulfil certain density and homogeneity requirements to ensure functional performance for hundreds of years. (Holt et al., 2010)

Bentonite can absorb nearly five times its weight of water and at full saturation it occupies a volume many times greater than its dry bulk weight. After a canister is placed in a deposition hole, the protective bentonite blocks will be exposed to natural water from the surrounding bedrock and thus will swell to fill the gap between the blocks and rock wall. The time for this swelling to occur depends on the amount of water exposure, either naturally from the bedrock or through an artificial wetting system. (Holt et al., 2010)

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16 In original design the backfill blocks and pellets used in the deposition tunnel should be manufactured from same material so that requirements set for backfill are completed.

17 It must be tested that manufacturing of pellets is possible with a selected material.
The purpose of artificial wetting tests was to investigate on the swelling behaviour of artificially wetted highly compacted bentonite buffer blocks used in deposition holes for nuclear waste containment. The target was to study how filling the gap between the buffer block and deposition hole with wetted materials could create homogeneous and rapid swelling pressure. It is expected that the thermal, mechanical, chemical and hydraulic properties of the whole deposition system would be better when the bentonite-rock gap is closed with a uniform material. In practice, it is desirable that the bentonite material has a volume increase sufficient enough to create pressure against the rock surface. Uniform bentonite buffer swelling into the gap would prevent rock spalling while lowering the risks of bentonite piping and erosion due to potential water flow. (Holt et al., 2013)

The results of this work indicate that artificial wetting could be a potential alternative to the buffer design (Holt et al., 2010). The results of this work showed that the use of gap filling bentonite material combined with artificial wetting can provide radial pressure over 100 kPa within the first days. Bentonite pellets were more effective at generating and maintaining swelling pressure, especially in the larger gap size of 50 mm corresponding to the current reference design by Juvankoski (2012). Bentonite slurries and sand were not as desirable as gap filling materials due to the extreme levels of radial pressure. Eccentric alignment did not cause detrimental effects or severely uneven generation of pressure. Saline groundwater resulted in less swelling pressure, both of the buffer alone and when in combination with pellet filling. (Holt et al., 2013)

18

**Customized Bentonite Pellets - Extrusion Manufacturing & Thermal Performance Properties**

KBS-3V concept vertical deposition holes are drilled in the bedrock to host spent nuclear fuel filled copper canisters embedded in bentonite clay. These vertical holes are made with certain manufacturing tolerances and also some extra clearance is needed for the actual bentonite buffer blocks installation process. Because of these facts there is always an unavoidable gap between the bedrock and compacted bentonite buffer. Posiva’s current reference design this gap has a nominal value of 50 mm with ±25 mm tolerances and the gap is to be filled with bentonite pellets. The pellets are to be poured into the gap, without vibration or consolidation to achieve a bulk density of 1075 kg/m³ (Juvankoski et al., 2012). Also the same kind of pellet filling material can be used to close the gaps between backfill blocks and the tunnel rock walls in the deposition tunnel section areas. The gap width between the backfill blocks and the theoretical tunnel wall/roof is designed to be 100 mm. The dry density of the backfill pellet fill shall be within the range of 900-1100 kg/m³ (Keto et al., 2012).

The benefit of using gap filler is that it improves the properties of the components within the whole system especially with respect to thermal and mechanical transfer. Gap fillers can secure better thermal conductivity and prevent spalling of the surrounding rock to the deposition hole. Also gap fillers might be needed to retain the adequate average density for example in cases when the gap itself might be enlarged or the density of the buffer blocks is lowered. Bentonite buffer blocks may need to be secured to stay in their places and also slowing down the possible water erosion (piping) may be needed for wet holes. The thermal conductivity of the pellet filling in the gap between the deposition hole rock surface and the bentonite buffer is one of the key parameters in thermal behaviour of the bentonite buffer. The disposed copper canister produces residual heat due to decay of radioactive products. The decay heat is conducted through the bentonite buffer and the pellet filling to the surrounding rock mass. If the thermal conductivity of the pellet filling is too low the surface temperature of the canister will rise too high and this may lead for example to a case where the spacing between adjacent deposition holes should be increased. (Marjavaara et al., 2014)

The main focus in the earlier studies has been the performance and properties of various commercially available bentonite pellets and granules (Marjavaara et al., 2011). The natural continuation step was to improve understanding on how to manufacture customized bentonite pellets for the chosen concept and

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18 These tests have not continued since 2013 but according to Juvankoski (the buffer design) it is possible that buffer pellets will be artificially wetted in new design (these are only rumours at the moment).
how customized bentonite pellets perform in practice during the nuclear repository construction process.

The work done using extrusion pellets showed that it was possible to manufacture pellets with higher water contents, with inert additions and production simulation runs could be made successfully for 6 mm and 8 mm sizes of bentonite pellet diameters. This means that it is possible to domestically produce in-house customized extruded pellets for future research and actual operations, with an achieved production rate of 130 kg/h. If the buffer design requires gap filling of a certain dry density value range of 820-1025 kg/m³ to be used, this can be achieved with several pellets (both roller compacted and extruded) as shown in this study. Higher water content values allow closer compatibility with the designed bentonite buffer water content. (Marjavaara et al., 2012)

The temperature gradients over the gap filling in the thermal conductivity tests varied between 2.0 – 4.1 °C/cm. The redistribution of water content in the pellet filling was observed in all tests. The redistribution of water content was more significant with higher thermal gradient and longer testing time. The measured thermal conductivity of the roller compacted pellets was 0.17 W/Km (dry) or 0.19 W/Km (water containing). The lower value was measured with the dried bentonite pellets. The used temperature gradient did not have a significant effect on the thermal conductivity values. The thermal conductivity of the extruded pellets was 0.17 W/Km which was 0.02 W/Km lower than with the roller compacted pellets of nearly the same water content. (Kivikoski et al., 2013)

The results of these experimental pellet studies are being used in the Finnish KBS-3V reference design work. Also the outcomes of the thermal conductivity task have been applied to long term safety related thermal modelling of the nuclear repository. The results are applicable for other waste management organizations and designs other than KBS-3V, where pellet filling of buffer and backfill areas is needed.

**Buffer Backfill Interaction Tests (BBI, 2015) – mock-up**

During recent nuclear waste repository related research work the need of sophisticated experimental laboratory equipment has been raised because full-scale experiments simulate repository performance rather well but they are time-consuming and expensive. On the other hand results of small scale laboratory tests cannot always extrapolate in full-scale situations. Therefore, new laboratory test equipment has been manufactured in 2014 at about 1/6 scale dimensions of the deposition area planned within Posiva’s spent fuel repository. The test equipment has been used in related studies for the KBS-3V concept, specific for early age bentonite performance. (Peura et al., 2016)

New test equipment was utilized to study the interaction between the buffer and backfill. The key areas of interest were different bentonite materials interactions during the nuclear repository construction phase and right after it. At the top of the deposition hole the bentonite buffer block, gap filling pellets, tunnel floor filling material and backfilling blocks are in close contact with each other. When water is introduced to this area several interactions between clay products will happen and also the behaviour of the water flow itself and its distribution may have several impacts. The possible heaving or uplift of the high density bentonite buffer to the lower density backfill may happen. The possible erosion of bentonite material may also occur. There is also uncertainty regarding water channel flow paths. More information about these topics is needed to support of repository construction planning and for the future larger scale in-situ EBS demonstration tests and long-term safety. One approach to provide more detailed knowledge about these matters is to perform scaled down laboratory studies. (Peura et al., 2016)

The goal of this test was to gain information about the effect of flowing water on bentonite blocks and pellets during and immediately after emplacement. Simulated deep ONKALO (saline) groundwater was made to flow through the deposition hole and tunnel filled with bentonite blocks and pellets. The hypothesis was that water may erode bentonite while flowing through the system. Out coming water was analysed for dry material and chemical composition, thus information about the erosion was gathered. (Peura et al., 2015)

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19 This is the first study in mock-up scale where behaviour of the buffer and backfill and their interaction are studied together.
The testing system hosted engineered barrier clay main components such as bentonite buffer, tunnel backfill and pellets. The scope of the test was to evaluate the clay barriers behaviour after introduction of water flow. One of the main topics of interest was the studies of water induced mechanical erosion of clay materials. The mechanical erosion was evaluated by measuring the water outflow and the dry material content of the water. Bentonite erosion as a function of time was obtained too. A scheme of the water flow system is presented in Figure 2. (Peura et al., 2016)

![Figure 2. Schematic view of the water flow system. Blue lines indicate the water flow, red dashed lines indicate for an automatic switch off system of the pump and black lines indicate a controlling system of the pump unit and an automatic sampler. (Peura et al., 2016)](image)

Saline water was made to flow through the testing system by a pump and sufficient inlet and outlet vents. The rate of water inflow was set to 0.1 l/min and it was channelled to the desired position in the system. The inflow rate was constantly monitored. Automatic sampler was used to collect the samples from the outflowing water at certain intervals. Beakers containing water were collected from the automatic sampler every working day and the samples were weighed and placed into an oven for evaporation. Evaporated samples were weighed for the dry material content. The swelling pressure inside the tunnel was monitored with strain gauges and pressure sensors. Two dial gauges were also attached to the sides of the middle tunnel frame unit to monitor the movement of the steel frame caused by the swelling pressure inside. (Peura et al., 2016)

After stopping the tests the system was opened and dismantled. During the dismantling a large amount of block and pellet samples were taken and the water content, bulk and dry density values and saturation degree were analysed. Test assembly and implementation and dismantling of the test are described more exact in the following chapters. The duration of the tests was limited to 2 months. (Peura et al., 2016)

**40% Scale Bentonite Buffer Test @ ONKALO**

A 40 % scale bentonite buffer test was commissioned to simulate geological disposal of nuclear waste according to the current reference design concept adopted in Finland. Two experimental vertical holes with a diameter of 800 mm were drilled into the tunnel floor at the testing place located 140 m underground. The depth of both of holes was about 3 m and the distance between holes was 4 m. Bentonite buffer blocks and a heating canister simulating nuclear waste were installed in the holes with monitoring equipment. During the test program one hole was artificially wetted. Thermocouples, relative
humidity transducer, total and pore pressure sensors, displacement and force transducers were installed to monitor the system behaviour. The bentonite buffer blocks were instrumented and preassembled to stacks in VTT’s laboratory and were then transported to the site in Posiva’s underground characterization facility (ONKALO) in Olkiluoto. (Hakola et al., 2015)

The goal of the 40% scale bentonite buffer test was to get information about methods and best practices for the installation, the behaviour of sensors and cables in nuclear repository conditions, and the early phase processes include distribution of the heat, the rate of saturation from the gap to the middle of the buffer block, the effect of watering, the possible piping and erosion, the swelling and the buffer uplift. (Hakola et al., 2015)

One experimental test hole was opened and sampled after two years of testing. The main goal of the dismantling of the selected hole was to study how to plan, perform and document the actual sampling and removal of the bentonite buffer in underground conditions. The work included studies and testing of the needed equipment, defining what were the best practices of bentonite sampling and how to handle and mark large number of samples in underground conditions and transfer them unchanged to the laboratory for assessments. The laboratory work included the determination of the water content and density distribution of the bentonite buffer blocks and the gap filling pellets. (Hakola et al., 2015)

The removal of the heating canister was found to be challenging. This had been anticipated and an additional plan had been crafted beforehand that needed to be taken into use. The dismantling plans of the electrical systems were found to be not detailed enough. Also there had been lack of documenting of certain electrical systems during installation phase two years earlier which affected the dismantling work. These topics should be assessed better in the following dismantling work of the on-going test hole experiment. There were underestimations of the actual amount of work needed in sample pre-processing in the laboratory. These estimations can be corrected based on the results of this work in the future experiments\textsuperscript{20}. The plans and procedures created during this study can be used as a baseline and can be further developed to better serve future work in ONKALO. (Hakola et al., 2015)

**POPLU, full scale demonstrations\textsuperscript{21}**

ONKALO is an underground bedrock research facility (URCF) excavated as part of the location studies performed in Olkiluoto in Eurajoki. The fact that URCF is a part of the final repository has both advantages and disadvantages from demonstration perspective. Naturally, research conducted at ONKALO provides an opportunity to innovate in the area of rock construction and to further develop final disposal techniques in realistic conditions. In addition, it has identified the areas where the construction of the final deposition tunnels would be most cost-effective. On the other hand the long-term safety requirements set strict limitations for demonstrations studies, as given example of POPLU studies shows.

The purpose of Posiva’s project called POPLU (full-scale demonstration of a deposition tunnel end plug) is to demonstrate the full-scale design, construction, performance and documentation of a tunnel plug in actual repository conditions of ONKALO. The plug constructed within POPLU will be the first plug constructed by Posiva.

The design basis for the reference deposition tunnel plug has been captured in the VAHA requirements management system (RMS) as a hierarchy of requirements. VAHA concentrates on post-closure requirements, and, therefore, the majority of the requirements on deposition tunnel plugs focus on how the deposition tunnel plug contributes to post-closure safety, i.e. by keeping the backfill in place during the operational phase and ensuring that the plug does not significantly affect the post-closure performance of the backfill. (White et al., 2014)

\textsuperscript{20} For example FISST = full scale demonstrations of whole EBS (starts 2017?).

\textsuperscript{21} This is still on-going project so results/conclusions are not available yet
The safety functions for POPLU are the same as those defined for the reference deposition tunnel plug in VAHA. Although the detailed requirements on POPLU are still under development, the conceptual design of the plug is based on a previous design developed by Posiva, and requirements on that conceptual design have therefore been used to identify the design basis for POPLU. (White et al., 2014)

The expected lifetime of the deposition tunnel end plug is 100 years, though the components will be in-place for thousands of years, and therefore the materials used in the plug should not impact the long-term performance and safety of the repository deposition. As the POPLU plug demonstration is being constructed in the future repository location of ONKALO, caution must be taken for materials used in experimental research and development. Even if POPLU materials will later be removed from the site prior to repository operation, they have the potential to leave traces to the surrounding groundwater and bedrock environment. Therefore, in the POPLU Experiment it was important to use the same materials as in real operational-phase plugs, to see if the initial state of the plug will be achieved. All materials used in POPLU were still subjected to Posiva’s review process and documentation for Foreign Material acceptance. (Holt, internal)

Foreign materials monitoring was introduced into ONKALO at the start of the construction in 2004. It covers the approval procedure for the materials used in the construction of ONKALO, bookkeeping of the materials used in the underground facilities, and monitoring of the effect of foreign materials on the groundwater. (Sacklén 2015) The processes of foreign material acceptance are described in Posiva’s Material Handbook, as also briefly summarized here.

The Material Handbook is a collection of documents providing information of the materials allowed in ONKALO. It includes separate instructions for the use of each material, material safety data sheets (MSDS) and other relevant information. These materials have been divided into two safety levels: Safety level A (the highest safety level) includes materials that could have an impact on long-term safety. Materials in safety level B have no detrimental influence on long-term safety according to present knowledge. Safety level A includes cementitious materials and additives, organic compounds, inorganic nitrogen compounds and other inorganic compounds. Safety level B includes metals and other materials. (Sacklén 2015)

The instructions for introduction of a new material are found in the material handbook. A new material can be approved for the use in ONKALO, if it is not harmful for long-term safety and it has been evaluated to be suitable for ONKALO conditions. Its functionality for ONKALO conditions must have been tested. For a material in the safety level A the disadvantages of its use must be less than the possible disadvantages if it is not used. (Sacklén 2015).

4.6 Summary and conclusions

This study aims to show with practical examples that the planning of nuclear waste disposal and practical implementation is an iterative process which is influenced by many different factors. The most directing factors are long-term safety requirements, undoubtedly. The long-term safety requirements may become more accurately or amount of them can rise when more knowledge is gathered but they do not change even if the technical implementation is changed. On the other, available techniques or economical requirements may set some limitations which are needed take into account in setting of long-term safety requirements.

22 Foreign materials are materials that are not part of the engineered multi-barrier system or the natural environment. They could have an impact on the long-term safety of the repository deposition and thus their use in ONKALO needs to be monitored.
Technical demands or requirements are divided into numerous subareas depending on whether it is in question of equipment or devices, working techniques or methods or other technical aspects. In addition, the right context must be noticed, such as component-specific requirements (buffer, backfill, closure etc.). For example the excavation of deposition holes and tunnels is carried out in different working methods which means that devices are different but also excavations tolerances (accuracy) are different based on used methods, not only justified long-term safety.

It is not possible to describe all technical requirements in this context because their amount is so huge but all experts who are working with them have to identify right requirement and interfaces which significantly affect his/her work. When approached to practical implementation the number of different technical demands and limitations increases and the safety aspects gets more dimmer even though it exists indirectly based on the design requirements. Sometimes this can be seen in practical work – prioritizing of numerous demands and requirements may be difficult. To make the prioritizing process easier, it would useful if every requirement is signified with its arguments. In this way it would be facilitate to evaluate the impact of changing the requirements or demands. In other words the status and justifications of requirements should be better documented not only for transparency reasons but also for historical reasons. The research of final disposal system takes decades so it is important to share all gathered knowledge for the younger generations.

Safety requirements affect the most directly to the design of EBS components. In design phase requirements become the actual practical implementation plan which can be verified by demonstration studies. Demonstration studies give valuable feedback and possible unrecognized actions of the functionality of EBS system. In future the scenarios where some components of EBS do not operate as planned should be tested also in practice.
5. Conclusions

This literature review has discussed the general concepts regarding safety and the Safety Case and as well as Safety Case in nuclear waste management. Some countries and their methods are looked at as examples and the structure and methodology regarding Safety Case were reviewed, as well as some practical limitations and imperfections to be found in existing Safety Cases.

This review is a part of the KYT2018 –project TURMET and should be read jointly with the literature review on Scenario Analysis by Edoardo Tosoni.

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Appendix 1: Illustration of the process used to develop the POPLU design basis and role of performance assessment. (White et al., 2014)