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POSIVA'S APPLICATION FOR A DECISION IN PRINCIPLE CONCERNING A DISPOSAL FACILITY FOR SPENT NUCLEAR FUEL

STUK's statement and preliminary safety appraisal

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ABSTRACT

In May 1999, Posiva Ltd submitted to the Government an application, pursuant to the Nuclear Energy Act, for a Decision in Principle on a disposal facility for spent nuclear fuel from the Finnish nuclear power plants. The Ministry of Trade and Industry requested the Radiation and Nuclear Safety Authority (STUK) to draw up a preliminary safety appraisal concerning the proposed disposal facility.

In the beginning of this report, STUK's statement to the Ministry and Industry concerning the proposed disposal facility is given. In that statement, STUK concludes that the Decision in Principle is currently justified from the standpoint of safety. The statement is followed by a safety appraisal, where STUK deems, how the proposed disposal concept, site and facility comply with the safety requirements included in the Government's Decision (478/1999).

STUK's preliminary safety appraisal was supported by contributions from a number of outside experts. A collective opinion by an international group of ten distinguished experts is appended to this report.

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STUK'S STATEMENT TO THE MINISTRY OF TRADE AND INDUSTRY

On May 26, 1999, Posiva Ltd submitted to the Government an application, pursuant to the Nuclear Energy Act, for a Decision in Principle (DiP) on a disposal facility for spent nuclear fuel from the Finnish nuclear power plants. The facility is proposed to be located at a site in the vicinity of the Olkiluoto nuclear power plant. That site is one of the four sites where detailed site investigations have been carried out.

In accordance with the Section 12 of the Nuclear Energy Act, the Ministry of Trade and Industry must obtain a preliminary safety appraisal from the Radiation and Nuclear Safety Authority (STUK) on the application for DiP. According to Section 14 of the Nuclear Energy Act, before making a DiP the Government shall ascertain, that no factors have arisen indicating lack of sufficient prerequisites for constructing the disposal facility so that it is safe; it shall not cause injury to people, or damage the environment or property.

The Ministry of Industry has requested for STUK's preliminary safety appraisal in a letter dated on June 24, 1999. STUK's position on the DiP application is given in this statement. The appended safety appraisal judges, how the proposed disposal facility, disposal concept and the disposal site comply with the safety requirements included in the Government Decision on the safety of disposal of spent nuclear fuel (478/1999, dated March 25, 1999). STUK's preliminary safety appraisal is supported by contributions from a number of outside experts. The statement by the Advisory Committee for Nuclear Safety, dealing with the DiP application and STUK's preliminary safety appraisal, is also appended.

The research, development and planning work related to disposal of spent fuel has progressed so that adequate basis exists for the consideration of the DiP

The preparatory works for spent fuel disposal has been conducted in accordance with the target schedule decided by the Government in 1983 and confirmed by the Ministry of Trade and Industry last time in 1995.

According to the Nuclear Energy Act, a DiP is required for a disposal facility of nuclear waste, and the explanatory memorandum of the Act clarifies that a disposal project should be submitted to the Government in an early phase. The Government would then decide whether the project is in line with the overall good of the society and accordingly, whether the continuation of the project is justified.

Research and development work on spent fuel disposal has been conducted in Finland during twenty years. Acquisition of the required further knowledge to ensure safety calls for research work at the planned disposal depths and conditions, in addition to the continuation of investigations made from ground surface.

Construction of an underground research facility implies large investments by the applicant, Posiva Ltd. STUK will not take a stand on the relation of these investments to the provision of Section 15 of the Nuclear Energy Act and Section

30 of the Nuclear Energy Decree, which limit the applicant's investments prior to the acceptance of the DiP. Nevertheless, STUK deems that continuous research work and construction of the underground research facility is important from the standpoint of safety.

In order to benefit from the further studies as much as possible, the underground research facility should be constructed at the site which is currently regarded as the most suitable one.

The investigations made thus far suggest that Olkiluoto is a suitable disposal site

In the review work made by STUK and its outside experts, no such geological factors emerged, which would support selection of some other site than Olkiluoto as the disposal site.

Given the other favourable characteristic of the Olkiluoto site (minimisation of transports, possibility of sea transports and already existing local infrastructure), STUK deems justified to focus the further investigations at Olkiluoto.

Operation of a disposal facility poses no significant safety hazards

The DiP application include a preliminary design of the disposal facility, consisting of an above-ground encapsulation facility and an underground facility with auxiliary facilities. In the planning work, much attention has been given to the determination of the safety related design bases for the facility. In STUK's view, these preliminary plans are appropriate and adequate in the present phase.

The disposal facility is predominantly based on proven technology. However, the manufacturing technology of the waste canister and its quality control require substantial further engineered development work.

The safety assessments for the operation of the disposal facility and for the transportation of spent fuel demonstrate sufficiently compliance with the safety requirements. Experiences from transports of spent fuel already exist abundantly, for instance in Sweden.

The safety of spent fuel transportation, encapsulation and disposal operations will mainly be based on inherently safe systems and these activities involve no potential for a severe accidents leading to environmental pollution.

Studies have strengthened the view that the safety objectives for the very long time perspective can be met

Reliable containment of the disposed spent fuel from the biosphere is intended to be ensured by means of a multiple barrier system. The long-term safety of disposal is justified by a safety assessment, which is supported by comprehensive studies made during two decades.

These studies, reviewed by STUK, and the publicly financed independent research work have strengthened the view that the established safety requirements can be met with considerable margins. The barriers are likely to retain their performance capability as long as required for the containment of the radioactive substances. The failure of barriers due to natural events, such as post-glacial rock displacement, or human actions, would neither lead to severe environmental consequences.

In the review reports and expert statements acquired by STUK, the suitability of the disposal concept is not questioned, or no other reasons for rejecting the DiP application are put forward.

More research and development work is needed for the support of the construction licence application

As clarified in the appended STUK's preliminary safety appraisal, the safety of disposal has not yet been demonstrated with the certainty as to be required for a construction licence application. Uncertainties and need for development are included in many of the relevant areas: the fabrication of waste canister, the long-term performance of engineered barriers, the characteristics of the proposed disposal site and the methodology of safety assessment.

The intensive research and development period of about ten years, proposed by Posiva, seems appropriate for resolving the open issues.

The proposed disposal concept provides a better approach for the containment of spent fuel than the other known alternatives

Alternatives for spent fuel management are discussed in STUK's statement (August 20, 1999) on Posiva's environmental impact assessment report. The statement concludes that disposal of spent fuel into bedrock is the alternative which involves least uncertainties in the Finnish circumstances.

An essential feature of Posiva's disposal concept is that in all parts it can be implemented by domestic efforts. This is consistent with the spirit of the Amendment of 1994 of the Nuclear Energy Act: our national responsibility is to take care of the waste that is produced in Finland.

Another strength of the concept is that the basic principles are applicable and necessary notwithstanding the type of nuclear waste. This fact minimises the need for further modifications in case that the international research and development work would result in new technical innovations.

The essential features of the concept include:

- The multiple barrier principle, ensuring almost unaltered level of safety if a single barrier failed;
- The elimination of a possibility of sudden and widespread releases of radioactive substances;
- The inaccessibility of the repository and the absence of need for monitoring in the post-closure period.

The disposal concept allows improvements, if substantial innovations would emerge in future

The international research and development work has not produced any revolutionary innovations during the two decades when the preparatory work for spent fuel disposal has been conducted in Finland. Nevertheless, the possibility of innovations cannot be completely ruled out and, for instance, new methods may emerge for decreasing the amount of waste to be disposed of or for further improvement of conditioning of the waste. The prospects offered by future development should be assessed regularly in the course of the planning work and particularly during the construction and operating licence processes.

Evidently, the innovations can be flexibly utilised for modifying of the disposal concept, if the safety or economy of disposal can thereby be enhanced. No foreseeable innovation will, however, eliminate the need for disposal of high-level waste.

Development of an essentially different concept for the management of spent fuel would be expected only with fast global growth of the use of nuclear energy

In most of the countries where nuclear energy is used, principally similar disposal facilities as the one proposed by Posiva, are under development. A common feature is waste disposal deep into bedrock, either spent fuel as such or the high-level reprocessing waste arising from the spent fuel. The engineered concepts vary depending on the selected conditioning process and the available geological media.

Currently, no such circumstances can be foreseen, which would steer this congruent development towards an entirely new track.

Development of essentially different kind of concepts for spent fuel management would require vast efforts. Allocation of resources in them would be justified only in case that rapid global growth of the use of nuclear energy would take place and the spent fuel, which is currently declared as waste, would be needed as raw material for fresh fuel. This kind of growth would call for adoption of new reactor types, which are able to utilise uranium more efficiently than the present ones. The development of such reactors has, however, substantially declined and is currently approaching the level of basic research.

Irrespective of the future development, some kind of deep repository would be needed, provided that the countries having reprocessing facilities will return to the producer the high-level waste arising from the process.

From the Finnish perspective, it is not prudent to wait for the potential global growth of the use of nuclear energy with the adoption of novel reactor types. Anyway, this kind of development is beyond the influence of Finnish organisations involved with nuclear energy.

A safe disposal solution has to be developed in the time period when the required technical and scientific expertise is still available

The best way to ensure the safety of the disposal facility and the reliable isolation of nuclear waste from the biosphere is to implement the research and development work, and the design, construction and commissioning of a disposal facility in a continuous process.

The DiP is an important interim milestone allowing to continue the preparatory works for disposal and to create preparedness for the application of a construction licence.

If the process aiming at commissioning of a disposal facility slows down substantially or will be suspended, it is evidently difficult to retain or reacquire the expertise needed for the continuation of the disposal project. This is true in particular if no continuation for the use of nuclear energy is foreseen after the shut-down of the existing nuclear power plants. It is likely that the decline in expertise will take place in near future, if the nuclear field will not offer attractive prospects for the young generation.

A delay in the preparatory works would involve, besides the falling-off in quality of the technical and scientific works, significant additional costs and need for increasing the funds for nuclear waste management.

Retrieval of the disposed waste and adoption of a new spent fuel management concept is feasible in each phase of the disposal program

The proposed concept allows the retrieval of waste in any phase of the disposal project, even in the post-closure phase. It is, however, highly unlikely that safety aspects would trigger the retrieval of disposed waste.

The most likely reason for retrieval of waste would be such global development that makes the disposed waste attractive as raw material for nuclear fuel, and the economical profitability of retrieval would then be a crucial factor.

STUK concludes that the DiP is currently justified from the safety point of view

In STUK's preliminary safety appraisal no factors were discovered, which would indicate lack of sufficient prerequisites for constructing the disposal facility in compliance with the safety requirements included in the Government Decision of March 25, 1999. Consequently the prerequisites for a DiP, included in the Nuclear Energy Act, are met from the nuclear safety point of view.

The DiP should be associated with a requirement for research and development work for ensuring adequately the safety of the disposal concept prior to the construction licence phase. The procedures for the planning and surveillance of the further preparatory works should appropriately be established by the Ministry of Trade and Industry in a decision pursuant to Section 28 of the Nuclear Energy Act.

The continuation of the disposal process in accordance with the DiP, or the need for the amendment of the DiP if required due to future world-wide development, can be considered in several phases. Such important decision-making milestones are the construction licence decision by the Government earliest in 2010, the operating licence decision by the Government earliest in 2020 and the decision on the closure of the disposal facility earliest in the mid-century.

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STUK's Preliminary Safety Appraisal

1 INTRODUCTION

On May 26, 1999, Posiva Ltd submitted to the Government an application, pursuant to the Nuclear Energy Act, for a Decision in Principle (DiP) on a disposal facility for spent nuclear fuel from the Finnish nuclear power plants. The facility is proposed to be located at a site in the vicinity of the Olkiluoto nuclear power plant. That site is one of the four sites where detailed site investigations have been carried out.

The disposal concept proposed in the DiP application has been the focus of research and development work performed in Finland during the past twenty years. The objectives for this work were originally defined in the Government Decision of 1983, including the following target schedule:

- Interim progress reporting in 1985 and 1992
- Preparedness for the selection of a disposal site by the end of 2000
- Preparedness for the construction licence application by the end of 2010
- Preparedness for the commencement of disposal operations as of 2020.

The preparatory works for spent fuel disposal were initially carried out by Teollisuuden Voima Ltd and since 1996 by Posiva Ltd. These preparations have so far progressed in accordance with the target schedule. Summary reports on disposal facility design, safety of disposal and disposal site selection were published in 1985, 1992 and 1996 and were subsequently reviewed by the Radiation and Nuclear Safety Authority (STUK). In 1993 an expert group, built up by the International Atomic Energy Agency (so called WATRP Review Team), evaluated the Finnish spent fuel disposal program (KTM 1994). In these reviews, the continuation of the disposal program was not questioned.

Subsequent to the DiP, if to be made, and prior to the submittal of the construction licence to the Government, Posiva intends to carry out a research and development program, which lasts about ten years and includes e.g. construction of an underground research facility at the selected disposal site.

In accordance with the Section 12 of the Nuclear Energy Act, the Ministry of Trade and Industry must obtain a preliminary safety appraisal from STUK on the application for DiP. This request is included in Ministry's letter dated on June 24, 1999.

STUK's preliminary safety appraisal is presented in this report. It is based on the Government Decision on the safety of disposal of spent nuclear fuel (478/1999, dated March 25, 1999). In the beginning of each topical chapter of this report, the relevant safety requirements included in the Government Decision, are quoted. Some of these requirements are specified mainly on the basis of STUK's YVL Guide under preparation. In each chapter, the safety requirements are followed by a judgement of how compliance with these safety requirements are demonstrated in the DiP application and in its most important reference reports.

The preliminary safety appraisal is supported by contributions from a number of outside experts. STUK has obtained statements on the safety aspects of the DiP application from the Technical Research Centre of Finland, Geological Survey of Finland, Laboratory of Engineering Geology and Geophysics/Helsinki University of Technology and Radiochemical Laboratory/Helsinki University. These statements are enclosed (Apps. 1–4). STUK also engaged an international group of ten distinguished experts to review Posiva's safety reports. A collective opinion by this expert group is given in Appendix 5 and Appendices 6–14 include nine topical reports by members of the same group. Furthermore, STUK has obtained judgements on the manufacturing technology and long-term performance of the copper-iron canister from the experts of the Manufacturing Technology/Technical Research Centre of Finland (Apps. 15–16).

2 WAY OF IMPLEMENTATION AND SCHEDULING OF DISPOSAL

Safety requirements

In accordance with section 6 of the Nuclear Energy Act, nuclear waste generated in connection with or as a result of the use of nuclear energy in Finland shall be handled, stored and permanently disposed of in Finland. In accordance with section 76 of the Nuclear Energy Decree when a decision is made on the principles that form the basis for the waste management obligation, the decision must be based on the premise that the nuclear waste can be transferred beyond Finland's jurisdiction for good or that it can be placed into Finnish ground or bedrock.

Section 7 of the Government Decision (478/1999) includes the following requirements:

The implementation of disposal, as a whole, shall be planned with due regard to safety. The planning shall take account of the decrease of the activity of spent fuel by interim storage and the utilisation of best available technology and scientific knowledge. However, the implementation of disposal shall not be unnecessarily delayed.

Disposal shall be planned so that no monitoring of the disposal site is required for ensuring long-term safety and so that retrievability of the waste canisters is maintained to provide for such development of technology that makes it a preferred option.

Subsequent to the selection of a disposal site, implementation of spent fuel disposal includes the following phases:

- Construction of an underground research facility;
- Construction of an encapsulation facility, auxiliary facilities and waste emplacement rooms;
- Encapsulation of spent fuel bundles and transfer of waste canisters into their deposition positions;
- Closure of emplacement rooms and other underground rooms;
- Post-closure monitoring, if required.

These phases, which can be partly parallel, should be scheduled and implemented with due regard to long-term safety. In doing so, the following aspects should be considered:

- Reduction of the activity and heat generation in waste prior to disposal;
- Introduction of the best available technique or a technique that is becoming available;
- Acquisition of adequate experimental knowledge of the disposal site and other components of the disposal systems affecting long-term safety;
- Potential surveillance actions related to ensuring the long-term safety or to non-proliferation of nuclear materials;
- Need for preserving the retrievability of disposed waste;
- Aim of preserving the natural features of the host rock and other favourable conditions in the repository;
- Aim of limiting the hazards and other burdens to future generations due to long-term storage of waste.

Retrievability means that even in the post-closure phase, it is possible to retrieve the waste canisters from the repository. In practice, retrievability will be limited to a time period when the engineered barriers provide an effective containment of the disposed radioactive substances. The disposal concept should be such that retrieval of waste canisters, if needed, is technically feasible with reasonable resources. Facilitation of retrievability or potential post-closure surveillance actions should not impair the long-term safety.

Compliance with the safety requirements

The proposed disposal concept is based on encapsulation of the fuel bundles, cooled down for at least 20 years, hermetically into composite containers of iron and copper. The waste canisters are emplaced into deposition holes where they are lined by bentonite clay (the buffer). The deposition holes are located in a wide network of tunnels at the depth of 400–700 m in bedrock. This disposal concept has been subject to extensive research and development work for about 20 years particularly in Sweden and Finland.

The overall disposal program put forward by Posiva, proposing commence the construction of the disposal facility and its commissioning around 2010 and 2020 respectively, is based on the target schedule decided by the Government in 1983 and confirmed by the Ministry of Trade and Industry last time in 1995. The shut-down of the disposal facility will be made, according to the DiP application, in 2040 at the earliest and in 2100 at the latest depending on how long nuclear energy will be generated in Finland. The application includes provisions for disposal of spent fuel from two additional nuclear power plants, besides the existing ones.

After the DiP, Posiva intends to construct an underground research facility at the selected disposal site. STUK and its international review team (App. 5) consider this as an appropriate step for obtaining experimental data for ensuring the safety and the suitability of the disposal site.

The later steps of Posiva's overall schedule should not be regarded as definite ones. With good reasons, construction and commissioning of the disposal facility may be delayed. The way of technical implementation of disposal is also broadly defined in the DiP application and may consequently be modified, as necessary.

Retrievability was originally not adopted as a design basis for the disposal concept. Posiva has reported on the feasibility of retrieval of waste canisters during the operational period and in the post-closure phase (Saanio&Raiko, 1999). The report concludes that retrieval is feasible with reasonable efforts and that the required technology is similar to that to be used in the construction and operation of the disposal facility.

Post-closure surveillance means monitoring of the disposed waste e.g. for ensuring the safety or for discovering any diversion of the nuclear materials. Such surveillance actions are not specified in Posiva's disposal plans. In case of crystalline, saturated host rock, it seems not possible to adopt a disposal concept that would allow direct monitoring of the disposed waste canisters, without impairing the safety. Consequently, surveillance of e.g. nuclear materials must be based on indirect methods intended to discover intrusion into the sealed repository.

If so desired in due course, some of the main tunnels and shafts of the disposal facility may be kept open after the operational period in order to monitor the repository and to facilitate retrievability, although the waste emplacement rooms should be backfilled without unnecessary delays.

In STUK's view, the scheduling and way of implementation of Posiva's disposal plan includes flexibility to such extent that the pertinent safety requirements can be taken into account.

3 DESIGN BASIS FOR THE DISPOSAL FACILITY

3.1 General design principles

Safety requirements

Sections 11–16 of the Government Decision (478/1999) include the following requirements:

To ensure the operational safety of the disposal facility and the long-term safety,

(1) proven or otherwise carefully examined and high-quality technology shall be employed;

(2) advanced quality assurance programmes shall be obeyed; and

(3) advanced safety culture shall be maintained in the design, construction, operation and closure of the disposal facility.

Operating experience from the disposal facility shall be systematically followed and assessed. For further safety enhancement, such actions shall be taken that can be regarded as justified considering operating experiences and the results of safety research as well as the progress in science and technology.

The systems, structures and components of the disposal facility shall be classified on the basis of their importance to the operational safety and to the long-term safety of the disposal facility. Their quality level and the inspections and tests required to ascertain and verify the quality level shall be adequate considering the importance to safety of the item concerned.

The functions at the disposal facility that are important to the maintenance of the integrity of fuel bundles and waste canisters, prevention of radioactive releases and to the radiation protection of the personnel shall be ensured.

Technical and administrative requirements and restrictions for ensuring the operational and long-term safety shall be set forth in the technical safety specifications of the disposal facility. Appropriate instructions shall exist for the operation, maintenance, regular in-service inspections and periodic tests as well as for transient and accident conditions. The reliable function of systems and components shall be ensured by adequate maintenance and by regular in-service inspections and periodic tests.

The personnel at the disposal facility shall be suitable for their duties, qualified and well trained. The competence of the personnel shall be maintained and enhanced through training programmes.

Compliance with the safety requirements

Some of the safety requirements included in the Government Decision are of such character that compliance with them should be judged not yet in the DiP phase but in the later licensing phases. Such requirements are in particular those concerning technical safety specifications, operational instructions, regular in-service inspections and periodic tests, qualification of personnel and follow-up of operating experience.

The disposal facility consists of an aboveground facility, which includes e.g. rooms for reception of spent fuel and for its encapsulation (the so called encapsulation facility), and an underground network of tunnels, where encapsulated spent fuel (waste canisters) will be emplaced.

The DiP application and the supporting Posiva's reports (Kukkola 1999b, Kukkola 1999c) include a preliminary design of the encapsulation facility. This design is, to a great extent, based on proven technology already in use, although in certain parts of the facility the technology is such that it does not exist in Finland or it is still under development.

Handling of spent fuel bundles in the hot cell's atmosphere lacks practical experiences in Finland but, on the other hand, at reprocessing plants handling of spent fuel bundles is by far more demanding than the planned operations at the encapsulation facility. Furthermore, in the past decade dry storages for spent fuel, where handling of fuel bundles resemble to that in Posiva's design, have come into common use.

The manufacturing, sealing and inspection technology of copper containers are currently subject to intensive research and development work both in Finland and Sweden and significant progress has been achieved during the past few years. Posiva has recently completed the fabrication of the first full-scale copper-iron container, but the quality of its micro structure does not meet the specifications to be required for an acceptable container. STUK will closely follow the future research and development work related to the disposal container and will report on any progress to be achieved in the field. STUK's definite judgement of the compliance with the quality specifications will be needed, at the latest, in the construction licence phase. Appendix 15 includes a detailed review of the present status of disposal container manufacturing technology.

The designs for the aboveground and underground parts of the disposal facility are presented in the DiP application and in Posiva's reports (Kukkola 1999b, Riekkola et al 1999). The construction engineering design is based on existing and proven technology. Compliance of the design of the underground facility with the respective safety requirements is discussed in Chapter 3.3. Transfer of the waste canister into the deposition hole with bentonite lining and backfilling of the emplacement tunnels will be tested in near future by full-scale experiments in the Swedish Äspö rock laboratory.

Posiva has a quality management system intended for ensuring the coherence in company's activities. It contains guidelines on three levels: the actual quality handbook (Posiva Oy 1999), procedural guidelines and working guidelines.

The quality handbook describes Posiva's organisation and the duties of its units and employees. The description of activities covers the strategic plan, branches of activity and contractual procedures between Posiva and its customers or subcontractors. The procedural guidelines give detailed guidance for the research, development and planning work. The working guidelines define the modes of operation and responsibilities in Posiva's own research and development work.

Posiva's quality management system is consistent with principles included in the international ISO 9001 quality management system. The descriptions included in the quality management system are appropriate and sufficient in detail and clarity. STUK deems that Posiva's activities have been conducted in compliance with the quality management system. STUK intends to audit Posiva's quality management system during 2000.

A high safety culture means that sufficient emphasis is given on safety in all organisations whose activities are involved with safety issues. Safety culture consists of organisations' modes of operation and individuals' attitudes.

STUK has observed Posiva's prevailing safety culture and its evolution on the basis of findings obtained in connection with the regulatory surveillance activities. Significant deficiencies have not been identified and STUK concludes that Posiva's safety culture complies with the safety requirements. As the disposal project proceeds, safety culture will be of growing importance to Posiva and its subcontractors and consequently, development of a more formal safety culture than thus far and its integration to the quality management system becomes necessary.

In the DiP application, Posiva has also presented views on the continuous safety improvement, safety classification of vital components and systems and on ensurance of safety functions. STUK has no objections to these preliminary plans but, even in this area, a more definite judgement of compliance with the safety requirements will be made in the construction licence phase.

On the basis of what has been said above, STUK deems that the reported preliminary plans for the disposal facility adequately fulfil the general design principles included in the Govenment Decision.

3.2 Prevention of releases and accidents

Safety requirements

Sections 17–22 of the Government Decision include the following requirements:

The dispersion of radioactive substances inside *the disposal facility as a consequence of handling of spent fuel shall be limited to the minimum. The released solid, liquid and particulate airborne radioactive matter shall be collected and treated as radioactive waste.*

Compliance with operational radiation protection constraints shall be ensured by means of continuous or regular monitoring, focused on the potential discharge routes at the disposal facility and on the activity concentrations in the surroundings of the disposal facility.

The formation of such spent fuel configurations that would cause an uncontrolled chain reaction of fission shall be prevented by means of structural design of systems and components.

The disposal facility shall be designed so that the likelihood of a fire is low and its consequences are of minor importance to safety.

The disposal facility shall be designed so that explosions that would jeopardise the integrity of spent fuel bundles, waste canisters, or the components or chambers containing radioactive substances, are reliably prevented.

The disposal facility shall be designed so that the impacts caused by potential natural phenomena and other external events are taken into account.

The Decision of the Council of State on the general regulations for physical protection of nuclear power plants (396/1991) shall apply, in accordance with its Section 12, to the disposal facility to the extent required by the degree of threat posed by unlawful activities to the nuclear facility concerned.

The Decision of the Council of State on the general regulations for emergency response arrangements at nuclear power plants (397/1991) shall apply, in accordance with its Section 10, to the disposal facility to the extent required by the degree of danger posed by the nuclear facility concerned.

Compliance with the safety requirements

The spent fuel to be handled in the disposal facility has cooled down for at least 20 years. Due to the long cooling time, the activity in spent fuel has decreased significantly and it contains only few nuclides which could be easily released in case of transient or accident. Because the residual heat of spent fuel to be handled is not particularly high and no high pressures or temperatures are needed in the handling chambers, it is very unlikely that major quantities of radioactive substances would be released inside the facility or into its environment.

If the disposal facility will be built at Olkiluoto, the encapsulation facility may be located next to the existing spent fuel storage building. Then the transfer of fuel bundles to the encapsulation facility would become simpler and it would be possible to limit further temperature transients in fuel and thereby also potential for radioactive releases.

The typical operational transients and accidents in the encapsulation facility and in the underground facility include malfunctions related to handling of a transport container, fuel bundle or waste canister, such as drop of a transport container of damage of a fuel bundle during handling in the hot cell. According to Posiva's preliminary designs and safety reports (Kukkola 1999a, Kukkola 1999b), limitation of consequences of accidents is based, for instance, on handling or storing of spent fuel bundles either in hermetic containers or in chambers with radiation protection arrangements. These arrangements include filtering of off-gases and sewage waters to the extent possible in order to prevent releases to the environment. The radioactive substances arising from the filtering systems and from decontamination of the handling chambers are recovered and treated as radioactive waste. The potential discharge routes of radioactive substances will be equipped with activity monitoring systems.

Posiva's reports (Rossi et al 1999, Kukkola 1999a) examine various transients and accidents and resulting radioactive releases and radiation exposures. On the basis of these analyses it can be concluded that the requirements for radiation protection and limitation of releases in the Government Decision can be met by employing the proposed, currently available technology.

The waste canister and the transport container are the most significant spent fuel concentrations in the disposal facility. The safety requirements call for prevention of criticality accidents by means of structural design of these components. For the waste canister, this has been shown in Posiva's report (Anttila M. 1999) and in a recent Swedish safety assessment as well (SKB 1999).

The criticality safety of the proposed spent fuel containers has convincingly been demonstrated by means of analyses and practical experience.

According to the DiP application, fire loads in the chambers of the disposal facility, where radioactive materials are handled, can be kept low. The causes of and consequences from fires will be limited by structural means, e.g. by fire cell, fire detection and alarm, fire extinguishing and smoke venting systems.

Explosives are handled in the underground facility. According to the DiP application, explosives are handled, stored, transported and used so that the likelihood of an accidental explosion is low and it would not be detrimental to radiation safety.

Design of the disposal facility takes into account the effects caused by potential natural phenomena and external events as described in STUK's Guide YVL 1.0 concerning safety criteria for the design of nuclear power plants. These events and phenomena include earthquakes, flooding and crashing aeroplane.

The DiP application explains that the arrangements for physical protection and emergency preparedness will be implemented in accordance with the pertinent Government Decision concerning the safety of nuclear power plants, taking into account that the consequences of potential accidents related to the operation of a disposal facility are less than those related to nuclear power plants.

As a summary of discussion above, STUK deems feasible to design and construct the disposal facility so that its operational safety is mainly based on inherently safe systems and in compliance with the pertinent safety requirements included in the Government Decision.

3.3 Design of the underground facility

Safety requirements

Sections 24–26 of the Government Decision (478/1999) include the following requirements:

At the planned disposal depth, blocks of bedrock with adequate size and intactness shall exist for the construction of the emplacement rooms. For the design of the emplacement rooms and for the acquisition of data needed for the safety analysis, the host rock shall be adequately characterised by means of investigations performed at the planned disposal depth.

The design, excavation, other construction and closure of the underground facility shall be implemented in the best manner with regard to retaining the characteristics of the host rock that are important to long-term safety.

Excavation works related to enlargement of the underground facility shall not be performed in the vicinity of disposed waste canisters and even otherwise the operations in the underground facility shall be designed with regard to efficiently prevent damages to waste canisters. Regarding underground excavation and construction works, transfers of rock masses or other comparable extensive transfers shall not be performed in the same areas which might simultaneously be used as transport routes for waste packages.

Compliance with the safety requirements

The repository has tentatively been envisaged to be located in the mica gneiss bedrock of the Olkiluoto site at the depth of about 500 m (Anttila P. et al 1999). The disposal tunnel network would be placed on one level and would accommodate an area of about 1 km^2, if the amount of fuel to be disposed of is 4000 tU. In case that the fuel amount is substantially higher, extension of the waste emplacement areas outside the focus area for site investigations would be necessary.

The view of the suitability of Olkiluoto's bedrock for disposal is based on the site investigations performed there during the past ten years. The proposed layout naturally sill stands for an outline of the repository and a more accurate location for the disposal tunnels cannot be determined until the completion of the investigations in the underground research facility, which would be constructed in the next phase of the site investigation program. The positions of the disposal tunnels are planned to be determined so that respect distances of at least 50 m are left to the major fracture zones included in the bedrock model. Posiva is developing a system for the classification of various weakness zones in bedrock.

A study on the constructability of the repository at the Olkiluoto site has been reported by

Posiva (Äikäs et al 1999). At Olkiluoto, the structural orientation and rock strength with respect to its stress state affect the design of the repository and may put limitations on the available disposal depth as well as on the shape and orientation of disposal tunnels (Apps. 12 and 13).

In the DiP application, vertical deposition holes bored at the bottom of a disposal tunnel are regarded as the primary method for emplacing waste canisters, but along with it, horizontal emplacement of waste canisters into tunnels is considered. The excavation of the disposal tunnels is planned to be made so that intersection of major fracture zones will be limited to the minimum practicable. Excavation is intended to be based on careful boring-blasting technique, a technique wherefrom extended experiences exist in Finland. Besides that, excavation by full-face boring technique will be considered, particularly in case that horizontal emplacement of waste canisters in tunnels will be adopted e.g. in order to limit the effects of rock stresses.

Disposal operations are planned to be performed so that after the excavation of a disposal tunnel, there will be no unnecessary delay between the emplacement of waste canisters and backfilling of the tunnel. This aims at limiting the need for rock injections and reinforcements in the tunnel and at prevention of transport of adverse substances, such as organic or oxidising agents, into the disposal tunnels. STUK's international review team points out that use of concrete and other potentially harmful materials should be limited when constructing the repository (App. 5).

Construction of disposal tunnels and emplacement of waste canisters into deposition holes is planned to take place congruently, but these activities would be separated by protective walls and respect distances would be adopted to dampen excavation vibrations. Materials to the construction and waste emplacement areas would be transported via different routes.

STUK deems that the preliminary design of the underground disposal facility and the plans for disposal operations take adequately into account the pertinent safety requirements included the Government Decision.

3.4 Implementation of control of nuclear materials

Safety requirements

Section 23 of the Government Decision (478/1999) includes the following requirements:

The design, construction, operation and closure of a disposal facility shall be implemented so that control of nuclear materials can be arranged in accordance with pertinent regulations.

Control of nuclear materials aims at verifying that nuclear materials are not transferred to nuclear weapons or other nuclear explosives and that nuclear materials are used in accordance with the licence conditions. Because disposal of spent fuel is a novel issue with respect to nuclear materials' control, the requirements for international control of underground facilities and related activities are still partly under development. The international treaties will not allow termination of nuclear materials' control even after the closure of the repository.

Control of nuclear materials call for verification of data on nuclear materials by measurements prior to disposal. The continuity of knowledge of the verified data shall be ensured until the closure of the repository.

The detailed implementation of nuclear materials' control will be defined in accordance with the existing national regulations and international agreements. These regulations have not yet established. International requirements for nuclear materials' control have been discussed and developed in particular within a multilateral international project (the so called SAGOR-project) under the leadership of the IAEA. IAEA has also established an expert group for the development and follow-up of nuclear materials' control related to final disposal. In the following, some requirements are tentatively given for the control of nuclear materials in spent nuclear fuel being disposed of.

Accountancy of nuclear materials, based on the respective accountancy at nuclear power plants, shall be continued for nuclear materials being handled or disposed of. All records shall be carefully deposited and secured.

In the encapsulation facility, the reported data on nuclear materials shall be verified with high accuracy, because it is the final place where individual fuel bundles are handled separately. Consequently, non-destructive measurements of nuclear material content must be feasible in the encapsulation facility to ensure that the reported data are correct and complete.

The transfer routes, handling processes and auditing of nuclear materials shall be planned so that continuity of knowledge can be ensured at each stage. If the continuity of knowledge is lost for a batch containing nuclear materials, the control measures shall be repeated for that batch. Closure on any waste emplacement room is not permitted until ensuring the continuity of knowledge for nuclear materials content in the waste canisters in that emplacement room.

Arrangements for nuclear materials' control and provisions for the positioning and use of the control instruments shall be considered in the design and construction of a disposal facility. The control program for a disposal facility and for the disposal operations shall include provisions for the verification of the design information and for the careful follow-up of the construction of the facility.

Compliance with the safety requirements

According to the DiP application, safeguards control would be based on accountancy of nuclear materials as well as on visual and technical measurement, surveillance and registration methods in the various stages of the disposal process. The technical principles for the control would include:

- Control of compliance with the planned and actual implementation of the facility (Design Information Verification);
- Accountancy of nuclear materials during the whole fuel cycle (Accountancy);
- Non-destructive assays for the verification of the quantity and quality of fuel (Non-destructive Assay, NDA);
- Continuous physical containment and surveillance of nuclear fuel to prevent losses and entanglements (Containment and Surveillance, C/S).

Control of nuclear materials is simplified because the fuel bundles are not disassembled at the encapsulation facility but are emplaced into waste canisters as such. However, a small number of capsules containing parts of previously disassembled bundles is likely to be encapsulated. Removal of the lid of a sealed canister and retrieval of fuel bundles will be required only if the canister is defective.

In the control of the implementation of final disposal, consideration shall be given to the modifications of design information due to the progress in host rock investigations and in underground construction works. Since a disposal tunnel will be backfilled as soon as possible after the emplacement of all waste canisters there, direct surveillance of waste canisters is not feasible thereafter and the control must be based on indirect methods.

STUK will actively participate in the development of international nuclear materials' control system. In STUK's view, control of nuclear materials can be implemented in accordance with the pertinent international treaties and national regulations as well as the principles and requirements given above.

4 PERFORMANCE OF BARRIERS

Safety requirements

Section 8 of the Government Decision (478/1999) includes the following requirements:

The long-term safety of disposal shall be based on redundant barriers so that deficiency in one of the barriers or a predictable geological change does not jeopardise the long-term safety. The barriers shall effectively hinder the release of disposed radioactive substances into the host rock for several thousands of years.

The system of barriers include both engineered and natural ones. Engineered barriers may consist of:

- Uranium matrix of low solubility, where most of the radioactive substances are incorporated;
- Hermetic, corrosion resistant and mechanically strong container, where the fuel bundles are enclosed;
- The chemical environment of waste canisters, which limits the dissolution and migration of radioactive substances;
- The backfilling material around waste canisters (the buffer), which has low hydraulic conductivity and which yields minor rock movements;
- The backfilling materials and sealing structures, which limit groundwater flow and transport of radioactive substances through excavated rooms.

Natural barriers may consist of:

- The intact rock around the disposal tunnels, which limits groundwater flow around waste canisters;
- The host rock where low groundwater flow, reducing and even otherwise favourable groundwater chemistry and retardation of dissolved substances in rock limit the mobility of radionuclides;
- The containment provided by the host rock against natural phenomena and human actions.

In the determination of the long-term performance of the system of barriers, consideration shall be given to sporadic deviations, as a consequence of which the performance targets are necessarily not met. Such deviations may be due to e.g. failures in the manufacturing or installation of engineered barriers, random variations in the characteristics of the natural barriers or their erroneous determination. The performance targets for the system of barriers as a whole shall be set so that the safety requirements are met notwithstanding the deviations discussed above.

The determination of the performance targets for the barriers shall be based on an assumption that the performance of a single barrier as a whole may be significantly lower than the respective target value due to some unpredicted phenomenon. 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 characteristics of the host rock can be assumed to remain in their present state up to an assessment period of several thousands of years. However, the effects of predictable processes, such as land uplift and disturbances due to the excavations and the disposed waste, shall be taken into account. 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 given above.

Compliance with the multiple barrier requirement

The performance of the barriers is significantly radionuclide dependent. Consequently, it is prudent to consider separately four groups of nuclides:

- Short-lived fission products, particularly strontium-90 and caesium-137;
- High mobility nuclides, particularly carbon-14, clorine36, selenium-79, palladium-107, tin-126, iodine-129 and caesium-135;
- Low mobility nuclides, particularly actinides and fission product technetium-99;
- Nuclides in the structural parts of fuel bundles.

Short-lived fission products

Strontium-90 and caesium-137 with half-lives of about 30 years dominate the activity of also the radiotoxicity of spent fuel for a few hundreds of years. If the engineered barriers provide an efficient containment for the waste for several thousands of years, as called for in the safety requirements, the short-lived fission products will completely decay during that time.

If, however, the containment provided by the waste canisters is not perfect even during the initial centuries, the fraction of activity in fuel's gas gap and grain boundaries will be crucial. Those radionuclides can be assumed to be released as soon as groundwater intrudes into the waste canister. This instant release fraction is estimated as 1% for strontiun-90 and 6% for caesium-137 in the TILA-99 safety assessment (Vieno and Nordman, 1999).

The solubility of caesium and strontium may be high but, on the other hand, their migration from fuel to rock through the so called near field is very slow and, according to the TILA-99 analysis, their activity will decay to a fraction of less than one millionth during migration. Furthermore, the transit time through geosphere is estimated to lower the activity of these nuclides almost to a similar fraction.

The considerations given above lead to a conclusion that three efficient barriers exist for the containment of short-lived fission products: waste canister, bentonite buffer and host rock. Substantial deficiencies in the performance of a single barrier seems not to jeopardise the compliance with the safety requirements, although uncertainties may affect significantly the release rates of strontium-90 and caesium-137.

High mobility nuclides

In TILA-99 analysis, the estimates for the instant release fractions of the high mobility nuclides vary between 1–12%. Because a typical annual release fraction from fuel matrix is estimated to remain between $10^{-6}…10^{-5}$, the instant release fraction will be crucial.

The relative activity of the high mobility nuclides with respect to the total activity of spent fuel is very low: for instance, after 1000 years it is about $5·10^{-5}$ and after 100 000 years about 10^{-3}. The total instant release activity is initially about 20 TBq, if the amount of spent fuel to be disposed of were 4000 tU.

High mobility nuclides have high solubility in groundwater and they are poorly retarded in bentonite and rock. Having a very long half-life, most of them will not decay significantly while migrating through the barriers. On the other hand, the peak release rate values (expressed e.g. in Bq/a) will substantially decrease due to dispersion phenomena. Likewise, the temporal dispersion in the loss of integrity of the waste canisters will provide a respective dilution for the instant activity releases of the high mobility nuclides.

The containment capability provided by the barriers is limited for the high mobility nuclides but, on the other hand, this implies that the activity releases arising from these nuclides would not crucially increase even if the performance of the barriers were much worse than assumed.

Low mobility nuclides

Low mobility nuclides dominate the activity and the radiotoxicity of spent fuel after a time period of a few hundred years. They have generally insignificant instant release fractions (nuclides in gas gap and at grain boundaries). Given the reducing conditions in the host rock, several barriers seem to exist for the containment of the low mobility nuclides: the long-lived container, incorporation into fuel of low dissolution rate, nuclide specific solubility limitations and slow migration through bentonite buffer and geosphere.

Because of the multiple containment for the low mobility nuclides, the activity releases and radiation exposure arising from these nuclides will, according to the TILA-99 analysis, be very low in comparison with those arising from the high mobility nuclides. This conclusion is valid even if the performance of a single barrier were substantially impaired. Only if there will be deficiencies in more than one barriers, like in case where a major fault intersects the repository, the radiation impact from the low mobility nuclides might become dominating.

Nuclides in the structural parts of fuel bundles

The metallic structural parts of fuel bundles contain both high and low mobility nuclides. The total activity of the high mobility nuclides is of the same order of magnitude as the respective activity in the fuel whereas the total activity of the low mobility nuclides is much less than that in the fuel. These nuclides are incorporated in a metal matrix wherefrom their release is estimated to take $10^3 \ldots 10^4$ years. A comparison with the nuclides in the fuel suggests that the nuclides in the structural parts of fuel bundles will generally have only a marginal effect on radiation impacts.

Conclusions

The proposed disposal concept seems to provide an efficient containment by the multiple barriers, if weight is given on the radioactive substances with the highest proportional activity in each assessment period. On the other hand, spent fuel contains also nuclides which will be released and migrated rapidly after the loss of integrity of waste canisters, but at any time their highest proportion of the overall activity will be about one thousandth.

Compliance with the requirement on engineered barriers

The most important engineered components of Posiva's disposal concept are the waste canister (copper-iron container enclosing fuel bundles) and the bentonite buffer around the waste canister. In addition, the hydrological and chemical conditions provided by these materials and the surrounding host rock are essential for the performance of the engineered barriers.

Copper, uranium dioxide fuel and bentonite are, on the basis of experimental evidence and thermodynamic analyses, very stable materials in the conditions that can be predicted to prevail in the repository.

The disposal concept proposed by Posiva aims at complete containment for very long times by means of the copper-iron container. The design basis for the waste canister is set so that corrosion of the copper container would take even millions of years in the repository environment. The design of the container takes into account, as far as possible, the geological changes due to e.g. future glaciations.

The required performance of the waste canister calls for fulfilment of high quality requirements concerning e.g. microstructure, faultlessness of weld seams and tolerances. Compliance with these requirements has not yet adequately proven because of the limited experiences on the manufacturing of massive copper containers. Both Posiva and the Swedish SKB have recently fabricated prototypes of full-scale waste canisters but their material properties have not yet reached the quality requirements to be called for a waste canister. During the past few years, there has been significant progress in the manufacturing technology of copper containers as well as in the electron beam welding technology of their seams and in related inspection technology. A more detailed review of waste canister fabrication technology is included in a report by the Technical Research Centre/Manufacturing Technology (App. 15).

Demonstration of the long-term integrity of the waste canister calls for exclusion of phenomena that might result in an early loss of integrity of the canister, such as creeping, local corrosion and stress corrosion. The importance of these phenomena are discussed in a review report by the Technical Research Centre/Manufacturing Technology (App. 16). It results in a general conclusion that further research is needed to judge the significance of these phenomena, which could jeopardise the integrity of waste canisters.

An essential feature with regard to the long-term integrity of waste canister is that the bentonite buffer retains the specified performance capability. Consequently, it has to fulfil stringent quality requirements with regard to e.g. material properties and compaction density. Essential properties for the mechanical stability include the ability of bentonite to yield rock movements and its ability to bear the waste canister having significantly higher density. To ensure the chemical and microstructural stability of bentonite, it is important to limit in the repository the peak temperatures and the quantities of materials containing readily dissoluble potassium and carbonate (Apps. 5 and 9). Future research is needed to get confirmed of these issues.

STUK deems that the engineered barriers included in Posiva's disposal concept have good potential for providing almost complete containment for radionuclides from the host rock for several thousands of years, as specified in the safety requirements. Getting confirmed of this calls for continuation of the research work and full-scale performance tests. Such activities are included in the planned research and development period subsequent to the DiP. Because the complete faultlessness of waste container fabrication, encapsulation and emplacement operations cannot be assumed, even the future safety assessments should take into account that a small fraction of waste canisters may loose their integrity substantially earlier than the specified minimum containment period.

5 SUITABILITY OF THE DISPOSAL SITE

Safety requirements

Sections 9 and 10 of the Government Decision (478/1999) include the following requirements:

The geological characteristics of the disposal site shall, as a whole, be favourable for the isolation of the disposed radioactive substances from the environment. An area having a feature that is substantially adverse to long-term safety shall not be selected as the disposal site.

The repository shall be located at a sufficient depth in order to mitigate the impacts of aboveground events, actions and environmental changes on the long-term safety and to render inadvertent human intrusion to the repository very difficult.

The characteristics of the host rock should be such that it adequately acts as a natural barrier, as specified in Chapter 4. Besides that, the characteristics of the host rock should be favourable with respect to the long-term performance of engineered barriers. Such conditions in the host rock as are of importance to long-term safety, should such be stable or predictable up to at least several thousands of years and thereafter the range of geological changes should be estimable.

Factors indicating suitability of a disposal site include.

- Proximity of exploitable natural resources;
- Abnormally high rock stresses;
- Anomalous seismic or tectonic activity;
- Adverse groundwater characteristics, such as lack of reducing buffering capacity and high concentrations of substances which might substantially impair the performance of barriers;
- Structural configuration of the host rock which is exceptionally difficult to interpret.

The disposal depth should be selected with due regard to long-term safety, taking into account at least:

- The geological structures and lithological properties of the host rock;
- The trends in rock stress, temperature and groundwater flow rate with depth;
- The dampening of the effects of aboveground natural phenomena, such as glaciation, and human activities.

To ensure that the effects of aboveground activities and phenomena will remain low enough, the repository should be located at the depth of several hundreds of meters.

Compliance with the safety requirements

The basis for the selection of disposal site was formed by a country-wide site screening completed in 1985 (Salmi et al 1985). The adopted site selection criteria for and the steps in the site selection process are described in Appendix 7 of the DiP application. Important factors in the selection of the investigation sites were in particular, besides the geological ones, local attitudes and landownership, because of being crucial to carrying out the siting process.

In the DiP application, the disposal facility is proposed to be located at Olkiluoto in Eurajoki municipality, one of the four sites where the site investigations have been completed. Posiva concludes that safe disposal would be feasible at all four sites and that the differences between the various sites are not very significant with respect to safety of disposal. Consequently, Posiva has founded the site selection decision mainly on other than geological factors.

The investigations made so far have revealed no factors which would indicate obvious unsuitability of any of the four sites, when judged on the basis of criteria given above. All investigation sites represent rock types which are quite common in Finland. The Olkiluoto site is reported to possess no particularly exploitable natural resources (Ilveskivi and Niini, 1985).

Lithological and structural characteristics

Interpretation of the bedrock structures has been made for all investigation sites. These interpretations, however, involve major uncertainties, because the available data on the orientations of fracture zones are not adequate (App. 3). The site scale bedrock structure model of each investigation site includes about thirty fracture zones. The bedrock structure models were adopted as a basis for a tentative identification of blocks of bedrock with sufficient intactness and other properties for hosting a repository. In the future investigations, the orientations and properties of structures inside these blocks will be studied.

The DiP application is based on an assumption that the emplacement rooms will be constructed at the depth of 400–700 meters. More than one bedrock blocks on different vertical levels may be needed to host the repository. At Olkiluoto, a repository for a fuel amount of 4000 tU is tentatively designed to be located in mica gneiss rock on one level at the depth of about 500 m (Anttila P. et al 1999).

A report on the geological characteristics and the constructability of the Olkiluoto site indicates that demanding rock reinforcement might be needed at the depth of more than 600 m (Äikäs et al 1999).

Hydrogeology

The data obtained at all investigation sites during several years indicate stable groundwater conditions and no major differences between the sites have been discovered. On the basis of hydrogeological investigations and interpretations, hydrogeological models have been prepared for the sites. A common feature for all sites is decrease of hydraulically highly conductive structures with depth. The hydraulic conductivity of intact rock seems also to decrease with depth.

The hydrogeological modelling is based on the bedrock structure models and on hydrogeological measurements. In the models of Olkiluoto and Hästholmen sites, the effect of land uplift has also been taken into account. STUK's international review team considered the geological investigation program being of high level, but criticised the reconciliation of the structural data with hydrogeological data being not as successful as it could be (Apps. 5 and 13).

Statistically fairly representative set of measurement data suggests that the hydraulic conductivity of intact rock at Olkiluoto would be slightly lower than the average of all investigation sites. Another favourable feature of the Olkiluoto site is the low hydraulic gradient and furthermore, it is predicted not to change significantly due to future land uplift. These factors imply that the groundwater flow rate at disposal depth of the Olkiluoto site seems to remain somewhat lower than that at the other investigation sites (Vieno and Nordman, 1999).

Groundwater chemistry

Posiva's investigations on groundwater chemistry are comprehensive and of high scientific quality, as concluded also in the reports by STUK's international review team (Apps. 5, 8 and 14). Identified deficiencies include lack of analyses on the chemical buffering capacity of the host rock with respect to geological changes (App. 9).

From the groundwater chemistry point of view, a major difference between the investigation sites is that the groundwater of the coastal sites (Olkiluoto and Hästholmen) turns from brackish to saline at the disposal depth whereas it is fresh at the inland sites (Kivetty and Romuvaara). The coastal sites have by far more complex groundwater chemistry than the inland sites. Furthermore, the groundwater chemistry at the coastal sites is in slow transition state due to land uplift, resulting in progressive decrease of salinity with time.

The studies made so far suggest that the effects of groundwater salinity on the safety of disposal are not significant, provided that the concentrations are not much higher than those of the Olkiluoto site at the disposal depth. The mobility of some nuclides may increase even in slightly saline groundwater. The performance of compacted bentonite will not deteriorate substantially until the salinity is a couple of times higher than at Olkiluoto's planned disposal depth. However, the salinity is growing rapidly with depth at Olkiluoto and may imply limitations to the available disposal depth.

Olkiluoto's groundwater chemistry at the disposal depth propose low groundwater turnover and long transit times (Anttila P. et al 1999). The apparent inconsistency with the results of groundwater modelling can be explained by the fact that groundwater chemistry gives a temporally and spatially integrated picture, whereas groundwater modelling is focussed on fast flow routes.

Rock movements

For the detection of slow rock movements, a network of GPS stations exist consisting of 12 country-wide stations and a local network at each investigation site. These measurements are intended to find the zones where horizontal or vertical rock movements are likely to appear due to land uplift, earth quakes or continental drift. The measurement stations have been in use only a few years and the measurement records are not yet sufficient for any definite conclusions.

Posiva has reported site specific studies, which include analyses of secondary displacements in the fracture network of the host rock, caused by an earthquake occurring in the vicinity of a disposal site (La Pointe and Cladouhos 1999). The results propose that such earthquake induced displacements might be a significant risk only in the post-glacial conditions, where the intensities and frequencies of earthquakes can be essentially higher than currently.

Studies on seismic histories of the investigation sites have also been made. At 100 kilometre radius around the Olkiluoto site, the earthquakes have been fairly small and infrequent during the observation history (Saari 1998).

The land uplift due to the latest glaciation is estimated to be currently 6 mm/a at Olkiluoto and 2 mm/a at Hästholmen at most (Miettinen et al 1999, Anttila P. 1999). The measurements indicate that the land uplift is continuously slowing down.

Conclusions

In STUK's view, no crucial differences exist between the investigation sites from the safety point of view. The investigations made thus far suggest that Olkiluoto is a suitable disposal site. The rock mechanical properties and the growth of groundwater salinity with depth may put some limitations on the available disposal depth. Similar conclusions were also taken by STUK's international review team (App. 5).

6 SAFETY ASSESSMENTS

6.1 Demonstration of the operational safety of the disposal facility

Safety requirements

Section 27 of the Government Decision (478/1999) includes the following requirements:

If compliance with the requirements for the operational safety of the disposal facility cannot be directly ascertained, it shall be demonstrated by experimental or computational methods or their combination. The computational methods shall be selected so that the detriment or risk likely to occur, with high degree of certainty, remains below the results of analyses. The applied computational methods shall be reliable and well validated for dealing with the events of interest.

Compliance with the safety requirements

Posiva has assessed the operational safety of the encapsulation facility and the underground disposal facility by a computational analysis. This analysis is based on the predesigns of the facilities and on experimental input data on the properties of spent fuel and its behaviour during transfer operations. (Anttila 1998, Rossi et al 1999, Suolanen et al 1999, Kukkola 1999a).

The radionuclide content and external radiation intensity of spent fuel was assessed at the Technical Research Centre (Anttila 1992, Anttila 1995 and Anttila 1998) by means of verified computed codes. The accuracy of the outcome of these calculations is good.

On the basis of the predesigns of the facilities, scenarios were drawn up for the analyses of the radiation exposures arising from normal operation, anticipated transients and potential accidents. In normal operation, the most likely source of radioactive releases would be leaking fuel rods. In transients and accidents, the releases would arise from fuel failures.

Radiation exposures for the various scenarios have been analysed (Rossi et al 1999). Occupational doses were calculated on the basis of the highest design basis radiation exposures inside the facilities. The exposure of individuals in the vicinity of the disposal facility is assumed to arise from radionuclides which penetrate the filters and are transported to atmosphere via ventilation stack. The variations due to weather conditions have been analysed probabilistically by employing an established computed code.

The analyses on the operational safety of the facilities are fairly simple with regard to the affecting physical phenomena. Nevertheless, the analyses involve uncertainties because of the approximate estimates for some input data, such as the number of leaking fuel rods, the activity fractions released from fuel and the shielding factors for exposed individuals. In the reported analyses, the selection of these input data targets to resulting doses which overestimate the really occurring exposure.

In STUK's view, the assessments of the operational safety of the disposal facility are, with regard to tentativeness of the plans, fairly comprehensive and are mostly based on methods which lead to overestimation of the really arising exposure.

6.2 Demonstration of the long-term safety

Safety requirements

Sections 28 and 29 of the Government Decision (478/1999) include the following requirements:

Compliance with 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. The safety analysis shall consist of a numerical analysis based on experimental studies and be complemented by qualitative expert judgement whenever quantitative analyses are not feasible or are too uncertain.*

Compliance with the radiation protection constraint given in Section 5 shall be demonstrated by assuming such a self-sustaining community in the vicinity of the disposal site that receives the highest radiation exposure. In addition to the impacts on man, potential impacts on species of fauna and flora shall also be examined.

The data and models introduced in the safety analysis shall be based on the best available experimental data and expert judgement. The data and models shall be selected on the basis of conditions that may exist at the disposal site during the assessment period and, taking account of the available investigation methods, they shall be site-specific and mutually consistent. The computational methods shall be selected on the basis that the results of safety analysis, with high degree of certainty, overestimate the radiation exposure or radioactive release likely to occur. The uncertainties involved with safety analysis and their importance to safety shall be assessed separately.

An assessment of the long-term safety of disposal of spent fuel should include at least:

- Description of the disposal system (waste canister, backfilling materials and sealing structures, excavated rooms, host rock and groundwater regime, disposal site) and definition of barriers;
- Analysis of the potential future evolutions of the disposal system (scenario analysis);
- Definition of the performance targets for the barriers;
- Required conceptual and mathematical modelling and the determination of the input data needed in these models;
- Analysis of the activity releases and resulting doses from radionuclides which are released from the waste, penetrate the barriers and enter to the biosphere;
- Whenever practicable, estimation of the probabilities of activity releases and radiation doses arising from unlikely disruptive events impairing long-term safety;
- Uncertainty and sensitivity analyses and complementary discussions on the significance of such phenomena and events which cannot be assessed quantitatively;
- Comparison of the outcome of analyses with the safety requirements;
- Documentation of the safety assessment.

Compliance with the safety requirements

In the DiP application, the demonstration of long-term safety of disposal is based on the TILA-99 safety assessment (Vieno and Nordman, 1999), which has been made by the Technical research Centre for Posiva. This assessment is not focussed on the Olkiluoto site alone but addresses all four investigation sites. TILA-99 assessment includes, in principle, all the elements which are required above for a safety assessment. Below, the adequacy and appropriateness of TILA-99 is discussed.

TILA-99 assessment includes no explicit scenario analysis, a lack which was identified also by STUK's international review group (Apps. 5, 7, 10 and 11). The authors of the analysis, however, have participated in international scenario analysis projects and utilised the knowledge received from these exercises in the definition of scenarios for TILA-99. In the context of the review work, no such features, events or processes of importance to safety were identified, which would be completely omitted in the TILA-99 assessment.

By nature, TILA-99 is a deterministic analysis, i.e. it employs point input data. The analysis is intends to be conservative, which means such selection of the conceptual models and input data that the outcome of the analysis, with great certainty, overestimate the radiation impact likely to occur. On the other hand, the uncertainty margins are extensive and consequently, absolutely conservative approach could not be applied for each model and data. Due to the variety of models and input data, it is difficult to judge the overall conservatism in the results of TILA-99.

The reference list of TILA-99 indicates that the assessment has aimed at utilisation of the best available experimental data and expertise. The researchers involved in the assessment have fairly good contacts to the most important research projects related to disposal of nuclear waste into crystalline bedrock. Posiva has agreements on co-operation and exchange of information with the foremost respective foreign organisations. STUK's international review group also deemed that TILA-99 assessment and its background studies are in line with the best international practice (App. 5).

The geological data in TILA-99 are, as far as practicable, derived from the results of Posiva's site investigations. They are generally representative for the bedrock of all four investigation sites but not specific to the proposed disposal site, Olkiluoto. Some of the data are, however, specific to saline groundwater (like at Olkiluoto and Hästholmen) or to fresh groundwater (like at Romuvaara and Kivetty). Furthermore, the effects of land uplift occurring at Olkiluoto and Hästholmen has been analysed.

In addition to what has been said above, the following deficiencies and simplifications in TILA-99, as identified likewise by STUK's review group (Apps. 5, 6, 7, 10 and 11), can be listed:

- TILA-99 is a single canister analysis. Consequently, the sensitivity in results arising from e.g. the variations in properties of rock surrounding the waste canisters cannot be easily deemed;
- The parametric values in TILA-99 are invariant in time, although the uncertainties involved in e.g. geological parameters increase with time. Consequently, the degree of conservatism in the assessment is deemed to decrease as the time period of interest grows;
- The scenario describing poor performance of the bentonite buffer is not representative for situations where the containment provided by bentonite might crucially deteriorate;
- In TILA-99, the analysis of radiation doses is simplified by applying only a well scenario, while the significance of other exposure pathways are estimated on the basis of other safety assessments. Besides that, the dilution factor assumed in the well scenario is substantially higher than that in some Swedish safety assessments (SKI 1996, SKB 1999);
- The radiation impact on living populations in the disposal site environment has been assessed in a quite limited extent. On the other hand, that subject still lacks experimental knowledge even internationally.

To counterweight the simplifications and uncertainties, TILA-99 includes a variety of sensitivity scenarios for the assessment of the effects of parameter variations on the results.

In STUK's view, TILA-99, along with certain other comparable safety assessments published in the past years (SKB 1992, Vieno et al 1992, SKI 1996, Vieno et al 1996, SKB 1999), can be adopted as a basis for the judgement of the long-term safety of the proposed disposal concept. During the forthcoming research and development period, the safety assessment methodology should be improved so that the deficiencies discussed above will be eliminated, and so that more site specific input data than currently can be introduced in the analysis.

7 RADIATION SAFETY

7.1 Operational safety of the disposal facility

Safety requirements

Section 4 of the Government Decision (478/1999) includes the following requirement:

The operation of a disposal facility shall not cause radiation exposure that could endanger occupational or public safety or could otherwise harm the environment or property.

The disposal facility and its operation shall be designed so that:

(1) as a consequence of undisturbed operation of the facility, discharges of radioactive substances to the environment remain insignificantly low;

(2) the annual effective dose to the most exposed members of the public as a consequence of anticipated operational transients remains below 0.1 mSv; and

(3) the annual effective dose to the most exposed members of the public as a consequence of postulated accidents remains below 1 mSv.

In the application of this Section, such radiation doses that arise from natural radioactive substances, released from the host rock or groundwater bodies of the disposal facility shall not be considered.

Compliance with the safety requirements

The planned disposal facility has good potential for limiting the operational radiation exposure to a low level, because:

- Only limited amounts of spent fuel with long cooling time is handled there at a time;
- The handling operations are fairly simple;
- No high temperatures or pressures are needed in the handling chambers.

The reference reports of the DiP application describe the normal operations and the possible transients and accidents in the disposal facility (Kukkola 1999a). They also include analyses on the radiation doses from normal operation as well as from transients and accidents to the most exposed individuals in the vicinity of the facility (Rossi et al 1999).

The assessment of the occupational radiation exposure is based on pessimistic assumptions, i.e. which target to overestimate the real doses. According to the analysis, most of the radiation exposure would arise from the receipt of transport containers. The annual collective occupational exposure would remain to a small fraction of that received at nuclear power plants, but in order to limit individual doses, exchange of workers during one year might be needed in the spent fuel reception activities. In STUK's view, a preferable approach is to plan the receipt of transport container so that significant reduction of radiation doses is feasible.

The doses to the most exposed individuals, due to releases from the normal operation, have been analysed probabilistically with respect to weather conditions. The results indicate that in the vicinity of the facility, the annual dose commitment would be about 0,01 mSv per year at most with 99,5 % confidence and 0,001 mSv per year at most with 95 % confidence.

Anticipated transients mean incidents with estimated average occurrence of less than once a year but having a significant probability to occur at least once during the operational period of the facility. The radiation doses to the most exposed individuals from such incidents has been assessed, with great certainty, to remain below one hundredth of the respective dose constraint of 0,1 mSv per year.

Postulated accidents mean incidents which have a low probability to occur during the operational period. According to the results of the analyses, the highest radiation doses from such accidents will, with high degree of certainty, be about one tenth of the respective dose constraint of 1 mSv per year, and only by assuming extremely unfavourable weather conditions these doses would approach the constraint.

With reference to discussion above, STUK deems that the operation of the disposal facility can be implemented in compliance with the safety requirements included in the Government Decision.

7.2 Safety of the transportation of spent fuel

Safety requirements

A DiP for spent fuel disposal need not to cover transportation of spent fuel from the NPPs to the disposal facility and it is neither included in Posiva's application. Because the safety of transportation is an issue which will probably emerge in the discussions related to the DiP, STUK will make below a brief review of the reports dealing with the safety of transportation.

The safety requirements for the transportation of spent fuel are included in the Act on Transportation of Dangerous Materials (719/1994) and the regulations pursuant to it. In addition, STUK's Guide YVL 6.5 addresses transportation of nuclear materials and nuclear waste.

The safety requirements for transportation are based on the approach that the transport container, in the first place, ensures the safety. Internationally standardised container types exist for transports of different kind of radioactive substances. For the transport of spent nuclear fuel, a B type container is required, which must qualify very demanding accident tests.

The transportation regulations set also constraints for the radiation dose rate outside the container in order to protect the transport personnel and other people in the vicinity of the container. These constraints are 2 mSv per hour on the surface of the container and 0,1 mSv per hour at the distance of one meter from the container. Furthermore, the regulations limit the outer surface contamination of containers in order to prevent spread of radioactive substances into the environment during transportation.

Compliance with the safety requirements

The health risks from transportation of spent fuel has been assessed in a report prepared by the Technical Research Centre (Suolanen et al 1999). The report considers transports from the NPP sites Hästholmen and Olkiluoto to the other sites being candidates for a disposal site.

The annual amount of spent fuel to be transported would be 110 tU on the average. The number of shipments would depend on the transport mode (road, rail or sea transport) and on the number of containers in one shipment.

The assessment of radiation exposure from normal situations is based on the maximum permissible dose rates during transports. The doses even to the most exposed members of the public will remain insignificant.

Assessed transients include a stopover of the transport, facilitating people to get close to the container, and releases of radioactive substances left on the outer surfaces of containers. The doses from these scenarios would remain low.

The report also includes analyses, which target to find out the consequences from severe damages to a transport package. Such accidents have not occurred thus far although thousands of transports of high-level waste have been made during more than 30 years. The results of these analyses indicate that, in the worst cases, temporary protection action for the public at the location of the accident might be needed.

Posiva's report makes also a comparison between the conventional traffic accident risks and the radiological risks due to transports. In conformity with some earlier studies on the same subject, the report concludes that conventional accident risk exceeds the radiological one.

In Posiva's reports, the safety of sea transports is discussed less comprehensively than the safety of land transports. If Olkiluoto will be the disposal site, spent fuel from the Loviisa NPP would probably be shipped to the disposal site by sea. In Sweden, experiences from sea transport of spent fuel over 15 years exist supported with comprehensive safety studies. Consequently, STUK deems that sea transport of spent fuel involves no such safety issues as would currently require further clarification.

Posiva's report and the international experiences from spent fuel transports over 30 years support the view that the transports of spent fuel to the disposal facility can be carried out in compliance with the international and national safety requirements.

7.3 Long-term safety

Sections 5 and 6 of the Government Decision (478/1999) include the following requirements:

In any assessment period, disposal shall not cause health or environmental effects that would exceed the maximum level considered acceptable during the implementation of disposal.

Disposal shall be so designed that as a consequence of expected evolutions, the radiation impacts remain below the constraints given in paragraphs 3 and 4.

In an assessment period that is adequately predictable with respect to assessments of human exposure but that shall be extended to at least several thousands of years:

(1) the annual effective dose to the most exposed members of the public shall remain below 0.1 mSv; and

(2) the average annual effective doses to other members of the public shall remain insignificantly low.

Beyond the assessment period referred to above, the average quantities of radioactive substances over long time periods, released from the disposed waste and migrated to the environment, shall remain below the nuclide specific constraints defined by the Radiation and Nuclear Safety Authority. These constraints shall be defined so that:

(1) at their maximum, the radiation impacts arising from disposal can be comparable to those arising from natural radioactive substances; and

(2) on a large scale, the radiation impacts remain insignificantly low.

The importance to long-term safety of unlikely disruptive events impairing long-term safety 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 specified in Section 5 above.

The long-term radiation protection objectives given above shall be followed in the design and implementation of a disposal facility. In accordance with the optimisation principle included in Section 2 of the Radiation Act and the continuous safety improvement principle included in Section 12 of the Government Decision, all practicable means shall be adopted for further reduction of radiation exposure, although the constraints were met.

The constraint for the most exposed individuals, effective dose of 0,1 mSv per year, will apply to a self-sustaining family or small-village community living in the vicinity of the disposal site, where the highest radiation exposure occurs. In the environs of the community, a small lake and a near surface well is assumed to exist. The exposure pathways to be considered include, as a minimum, the use of contaminated waters as household and irrigation water and the food chains in contaminated watercourses.

In addition, the safety assessment shall address the average effective annual doses to larger groups of people, who are living at a regional lake or a coastal site and are exposed to the radioactive substances transported into these watercourses. These average doses shall be, depending of the number of exposed people, not more than one hundredth – one tenth of the constraint for the most exposed individuals.

The nuclide specific constraints for the release rates of disposed radioactive substances to the environment, referred to in Section 5 of the Government Decision, will be specified in STUK’s YVL Guide. In the review of TVO-92 safety assessment, reported by STUK in 1994, the following nuclide group specific constraints were adopted:

- 0,1 GBq/a for radium-226, thorium-229, protactinium-231, uranium-238, plutonium-239 and -240, neptunium-237 and americium-243;
- 1 GBq/a for chlorine-36, selenium-79, niobium-94, tin-126, iodine-129 and caesium-135;
- 10 GBq/a for carbon-14, nickel-59, zirconium-93, technetium-99 and palldium-107.

The constraints need to be reconsidered in the light of the latest biosphere analyses (e.g. SKB 1999), which may imply changes in the constraints but probably not more than by a factor of ten. The activity releases can be averaged over 1000 years at most.

Typical unlikely disruptive events impairing long-term safety, referred to in Section 6 of the Government Decision, include:

- Boring a deep well at the disposal site;
- Core-drilling hitting a waste canister;
- Rock movement damaging a number of waste canisters.

The importance to safety of any such incidental and unintentional event should be assessed and whenever practicable, the resulting annual radiation dose or activity release should be calculated and multiplied by the probability of its occurence. This expectation value should be below the radiation dose or activity release constraints given above. If, however, the resulting individual dose might imply deterministic radiation impacts (dose above 0,5 Sv), its annual probability of occurrence should remain below 10^{-6}.

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 land and aquatic populations in the disposal site environment. 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 and economically significant animals and plants as well as domestic animals shall not be exposed detrimentally as individuals.

Compliance with the safety requirements

Radiation dose constraints

Because the technical barriers are required to provide an efficient containment for several thousands of years, releases of radioactive substances into biosphere in this time period could take place only if, due to e.g. fabrication or installation defects, a small number of waste canisters would prematurely loose their integrity. This kind of situation is described by the “pinhole” scenario of TILA-99 safety assessment.

If the other barriers performed as assumed, the highest radiation doses would arise from the high mobility nuclides, such as iodine-129, caesium-135 and carbon-14. The maximum dosed during the first 10 000 years would be less than 10^{-9} Sv per year for each failed waste canister.

The radiation doses from the “pinhole” scenario are not sensitive to most of the uncertainties related to the performance of barriers. Substantially higher radiation doses would arise only from extreme scenarios, for instance if an initially leaking canister, high groundwater flow around the waste canister and saline groundwater chemistry occurred simultaneously.

However, TILA-99 analysis pays inadequate attention to multiple failures with potential causal relation. A scenario like this is, for instance, the growth of an initial “pinhole” due to corrosion of iron and the deterioration of bentonite’s containment capability because of the outburst of corrosion gases or swelling of corrosion products. If this kind of scenario is assumed to take place during the first thousands of years, the maximum radiation doses would amount to 10^{-7}…10^{-6} Sv per year and failed canister according to the TILA-99 analysis. On the other hand, a recent Swedish safety assessment (SKB 1999) concludes that in spite of the “pinhole”, releases of radioactive substances would not occur until after 200 000 years.

TILA-99 include no analyses on radiation doses to larger groups of populations. Instead of that, it refers to Swedish safety assessments (SKB 1992, SKI 1996), which compare the maximum radiation doses from the well scenario with those arising from the coastal scenario. The “coastal” radiation doses, due to consumption of sea fish, would be below the “well” doses at least by a factor of 100 for the most critical nuclides (carbon-14, caesium-135, tin-126 and selenium-79) and by a factor 1000 for the other nuclides. Similar results were also obtained in the safety assessment for the disposal facility of low and intermediate level waste at Olkiluoto (Vieno et al 1991).

Activity release constraints

The activity release constraints are applied within the time period beyond several thousands of years, when major climate changes and as a consequence, significant geological changes are likely. Although the design basis lifetime of copper container is millions of years, the uncertainties related to its integrity will increase due to said geological changes and, on the basis of the principle of conservatism, it is reasonable to assume the loss of integrity of container much earlier than the design basis lifetime.

In order to judge the compliance with the release rate constraints, an illustrative and apparently conservative assumption is made that the copper-iron containers start to loose their integrity at 10 000 years and after 50 000 years the barrier function of all containers has been lost The resulting activity releases can be assumed on the basis of the “disappearing canister” scenario of TILA-99. The highest releases to the biosphere and the percentages of the respective constraints for the most critical nuclides would be as follows:

Carbon-14:	0,5 GBq/a,	5% of constraint
Cloride-36:	0,1 GBq/a,	10% of constraint
Tin-126:	0,3 GBq/a,	30% of constraint
Iodine-129:	0,01 GBq/a,	1% of constraint
Caesium-135:	0,04 GBq/a,	4% of constraint.

In the scenario assumed above, the difference between fresh and saline groundwater is not significant. Even more generally, the release rates are not particularly sensitive to the uncertainties in the performance of a single barrier, because the critical nuclides belong to the high mobility group. Only in case that the performance of more than one barriers would not meet the performance targets, the releases of the low mobility nuclides might become dominating.

Disruptive events

TILA-99 contains no explicit analysis on radiation doses arising from a deep well (e.g. 300 m) bored in the vicinity of the repository, but these doses can be approximated on the basis of the results of the shallow well scenario included in TILA-99. In a deep well, dilution would probably be much less than in a shallow well but, on the other hand, the likelihood of the existence of the deep well at the moment of interest would be low. Consequently, the expectation value of the deep well dose would not be substantially higher than the shallow well dose.

The consequences from a deep boring or core drilling has neither been analysed in the TILA-99 report. Scenarios of this type have been analysed in earlier Finnish and Swedish safety assessments (Vieno et al 1985, SKB 1999). The results indicate typically that the resulting radiation doses, e.g. to workers handling the drill core, can be high. On the other hand, the probability of hitting a waste canister by drilling, estimated on the basis of the current frequency of deep drillings, remains to the magnitude range of 10^{-7} per year. Thus, the expectation value of the radiation impact would be by far below the constraint.

TILA-99 includes an outline of a scenario, where a major rock displacement breaks a number of waste canisters. That scenario also assumes the loss of containment by bentonite and of unfavourable parameter values for groundwater flow and chemistry. A rock displacement is likely to occur in post-glacial conditions and consequently TILA-99 assumes it to happen at 30 000 years from now on. Recent studies on climate changes and ice ages, however suggest the glaciation to approach considerably later than TILA-99 assumes (Forrström, 1999, SKB 1999).

The resulting dose from the rock displacement scenario is 0,03 mSv per year for each broken waste canister and the dominant nuclide is plutoniun-239. Given that in each displacement up to some tens of waste canisters might be broken, the maximum dose would be about 2 mSv per year. If the post-glacial conditions are assumed to prevail not until after 50 000 years, the dose would be about half of the value given above.

Quantitative estimation of the likelihood of a rock displacement intersecting the repository is highly uncertain on the basis of the present knowledge, although studies on the post-glacial displacement in Fennosscaninadian have been made during the past years (Kuivamäki et al 1998). If the repository is located appropriately, the likelihood of a such displacement can be deemed to be small even in the forthcoming post-glacial conditions. Consequently, the expectation value of radiation dose can be estimated to remain well below the constraint of 0.1 mSv per year.

Protection of other living nature

The international principles and radiation exposure criteria for the protection of the other living nature are still under development (IAEA 1999).The current view is that living species shall be protected as populations, with the exception of rare or economically significant species and domestic animals. The best currently available scientific knowledge suggests that a radiation dose of 0,1 mGy per hour (about 800 mGy per year) to a fraction of individuals in a healthy population will not affect detrimentally this population (UNSCEAR 1996). Said radiation dose is over thousand times more than the constraint for the most exposed people due to disposal and more than hundredfold in comparison with the natural background radiation. It is consequently evident that the established radiation protection requirements ensure adequate protection of living populations and preservation of biodiversity.

Presumably protection of man will ensure adequate protection of rare and economically valuable species and domestic animals as well. The potential impacts of waste disposal will also be limited because the radiation exposure is concentrated in a fairly small area.

TILA-99 analysis refers to certain Canadian and Swedish studies, where the radiation exposure of plants, mammals, birds and fishes is assessed. All these estimates remain significantly below the potentially detrimental exposure levels.

The ethical principles and criteria for the protection of plants and animals and related methods for dose assessment will be extensively studied within various international organisations (ICRP, IAEA, EU) during the coming few years. Protection of other living nature is adequately considered in the present safety case, in spite of the limited discussion on the subject in the DiP application and TILA-99 analysis.

Conclusions

On the basis of the safety reports referred to in the DiP application, the appended review reports and its own review work, STUK deems that the proposed disposal concept and site comply with the requirements for long-term safety. This conclusion calls for that the performance of the system of barriers as a whole is not crucially lower than that assumed in the safety assessment. In particular, the performance of the engineered barriers during the first thousands of years is important to safety. To get confirmed of the safety, comprehensive studies are needed during the forthcoming research and development period.

8 SUMMARY

The disposal concept proposed by Posiva is consistent with the provisions of the nuclear energy legislation on nuclear waste management. Posiva's implementation program for disposal complies with the target schedule decided by the Government in 1983 and confirmed by the Ministry of Trade and Industry last time in 1995. In STUK's view, the scheduling and way of implementation of Posiva's disposal plan includes flexibility to such extent that the pertinent safety requirements can be taken into account.

The disposal facility, including the aboveground encapsulation facility and the underground repository with auxiliary facilities, has been preliminarily designed with focus on the determination of the safety related design bases for the facility. STUK deems these preliminary plans appropriate and adequate in the present phase.

Safety assessments have been prepared for the operation of the disposal facility and the transportation of spent fuel to the facility. In STUK's view, these assessments demonstrate sufficiently compliance with the safety requirements. Transportation, encapsulation and emplacement of spent fuel will be based mainly on inherently safe systems and involve no potential for a severe environmental accidents.

The long-term safety of disposal is planned to be based on a system of multiple barriers. These barriers are natural, like the bedrock, and engineered structures, like the waste container. The proposed disposal concept seems to provide an efficient containment by the multiple barriers, if weight is given on the radioactive substances with the highest proportional activity in each assessment period.

STUK deems that the engineered barriers included in Posiva's disposal concept have good potential for providing almost complete containment for radionuclides from the host rock for several thousands of years, as specified in the safety requirements. Getting confirmed of this calls for continuation of the research work and full-scale performance tests. Such activities are included in the planned research and development period subsequent to DiP.

The detailed site investigations have been completed at four sites. STUK has evaluated their suitability for disposal and judges that no crucial differences exist between the investigation sites from the safety point of view. The investigations made thus far suggest that Olkiluoto is a suitable disposal site. The rock mechanical properties and the growth of groundwater salinity with depth at Olkiluoto may limit the most suitable disposal depth to less than 700 metres, which is the maximum depth specified in the DiP application.

The long-term safety of disposal is justified by a safety assessment, which is supported by studies made during the past two decades. In STUK's view, that safety assessment, along with certain other comparable assessments published in the past years, can be adopted as a basis for the judgement of the long-term safety of the disposal concept proposed in the DiP application.

In the light of the safety reports referred to in the DiP application and of the review of the application, STUK holds the view that the proposed disposal concept and site have good potential for complying with the requirements for long-term safety. Getting confirmed of the long-term safety requires, however, substantial further research and development efforts, which are planned to be carried out during the forthcoming period e.g. at the selected disposal site.

STUK's preliminary safety appraisal is supported by a number of expert review reports and statements, which are appended to this document. In their statements, four research institutes which have participated in the publicly financed waste management research program state that, in the domain of their expertise, no facts have appeared indicating that the proposed disposal concept would not meet the safety requirements. STUK's international review group deems that from the safety point of view, the DiP should not rejected. STUK's own review work has neither revealed any facts leading to a conclusion that the provisions of the Government Decision (478/1999) are not fulfilled.

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Appendix

STUK External Review Group Consensus Report

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F.P. Glasser, B. Grundfelt, D.P. Hodgkinson,
J.A. Hudson, A.G. Milnes, K. Pers, D. Read

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1 INTRODUCTION

Posiva has recently published a performance assessment TILA-99 [Vieno and Nordman, 1999] which, together with supporting research and site characterisation reports, provides support to their application to the Finnish Council of State for a Decision in Principle regarding a final disposal facility for spent nuclear fuel [Posiva, 1999].

The Decision in Principle will be decided upon in the year 2000 on the basis that:

- Disposal is of overall benefit to Finnish society;
- A host municipality is in favour;
- Such facts have not arisen which show that there are not sufficient prerequisites to construct a safe nuclear facility.

The municipality of Eurajoki, which contains the Olkiluoto site, has indicated that it is willing to accept a repository.

The Finnish Radiation and Nuclear Safety Authority (STUK) has formed an External Review Group in order to assist with their response to the Decision in Principle application. The Group was selected so as to cover the range of scientific and technical disciplines that form the basis of the Posiva submission. The members of the group are:

- Mick Apted, Monitor Scientific, USA;
- Neil Chapman, QuantiSci, UK;
- Shaun Frape, University of Waterloo, Canada;
- Fred Glasser, Aberdeen University, UK;
- Bertil Grundfelt, Kemakta Konsult, Sweden;
- David Hodgkinson, Quintessa, UK;
- John Hudson, Rock Engineering Consultants, UK;
- Geoffrey Milnes, GEA Consulting, Sweden;
- Karin Pers, Kemakta Konsult, Sweden;
- David Read, Enterpris, UK.

This report provides a consensus of the views of the External Review Group at their meeting at STUK on the 9th and 10th September 1999. The individual, detailed review comments of the Group are also documented separately

The review considers the following questions:

- First, is the information presented, including the TILA-99 performance assessment and supporting research, adequate for the Decision in Principle?
- Given the results of this first evaluation, are there areas where Posiva should develop their performance assessment methodology and research programme in preparation for the Preliminary Safety Assessment Report (PSAR) in the year 2010?
- What are the implications for STUK?

The structure of this consensus statement is as follows. Section 2 considers the Posiva safety concept and research within an international context. Some specific issues arising from TILA-99 and supporting research are considered in Section 3, whereas the choice of Olkiluoto as the preferred site is discussed in Section 4. On the assumption that the Decision in Principle is positive, Section 5 makes some recommendations for the Posiva programme over the ten-year period to the Preliminary Safety Assessment Report (PSAR). Finally, some implications for STUK are presented in Section 6.

2 THE SAFETY CONCEPT— INTERNATIONAL CONTEXT

The Posiva *safety concept*, which envisages the disposal of spent nuclear fuel in high integrity containers surrounded by low-permeability bentonite within a deep geological repository, conforms to an international consensus for the final management of highly radioactive spent fuel wastes [NEA, 1999]. In particular, the Posiva approach is based on and follows closely the KBS-3 concept [SKBF/KBS, 1983] that has been studied intensively for almost two decades without any major problems emerging. With this background, the External Review Group is of the opinion that the Posiva *safety concept* is fundamentally sound and is based on high quality science and engineering which is in keeping with the best international practice.

Specifically, the External Review Group endorses the following safety features, which form part of the TILA-99 performance assessment.

There is a general consensus that the copper/iron canister acting in conjunction with the bentonite buffer could provide containment of radionuclides for at least hundreds of thousands of years provided that:

- Canisters are free from initial defects;
- Near-field geochemical conditions (Eh, pH, anions, salinity) remain stable following sealing and resaturation of the repository;
- Transport of corrodants in the buffer remains diffusive;
- The near-field rock remains mechanically stable.

Furthermore, given a stable and reducing near-field environment there is agreement that following degradation of the canister and consequent canister failure, radionuclides will be released only slowly into the geosphere by virtue of the:

- Slow dissolution rate of the spent fuel matrix;
- The low solubility of many key radioelements under reducing conditions;
- The limited abundance of highly soluble radionuclides in the inventory;
- Slow diffusion and retardation in the bentonite buffer;
- Physical filtration by the bentonite buffer acting to prevent any radio-colloid migration from the spent fuel to the host rock;
- The likelihood of limited failed canister surface area in any initially defective canisters.

The geosphere is expected to provide a generally stable chemical and mechanical environment protecting the near field, in addition to providing physical isolation of the waste from the future activities of people. The potential role of the geosphere in retarding the migration of radionuclides is likely but difficult to prove. It is noted that TILA-99 considers a number of variant scenarios that take little credit for this potential barrier function. Finally, radionuclides reaching the biosphere may be subject to net dilution before entering the food chain or appearing along alternative exposure pathways.

It is also noted that the overall system of safety barriers has a significant element of resilience to dynamical changes and uncertainties. Thus, it is the consensus of the External Review Group that the long-term, safe isolation of spent nuclear fuel should be robustly assured by the Posiva design over a wide range of anticipated future conditions.

As regards the presentation of the safety case, Posiva are to be commended on the aims of TILA-99 to be robust, transparent, traceable and reproducible. Posiva has also successfully conducted a policy of incrementally improving the safety assessment through iteration, for example from TVO-92 [Vieno et. al., 1992] and TILA-96 [Vieno &

Nordman, 1996] to the present report. TILA-99 represents a competent and pragmatic approach to safety assessment, and reviewers familiar with the performance assessment approach agreed that it is a well-presented report that generally outlines the issues clearly and in a sufficient level of detail to enable the reader to follow the reasoning easily. The supporting reports are also clear and well written, and, in some areas (particularly site hydrochemistry), are at the leading edge of understanding and interpretation. This has resulted in safety assessment documentation that is in line with the best international practice [NEA, 1997]. However, the documentation is not particularly accessible to non-specialists, including the general public, who need to be convinced of the arguments. Thus there is a case for producing a further report directed to this audience.

In summary, comparison with international practice has not revealed any reasons why a safe repository for spent fuel could not be constructed in Finnish bedrock and, from this point of view, it can be recommended that a Decision in Principle is taken to continue with the repository development programme. However, as discussed in the following sections, the detailed review has highlighted a number of issues where, granted a Decision in Principle, Posiva would need to focus effort if the programme is to go forward satisfactorily.

3 TILA-99

Notwithstanding the overall positive view of the Posiva safety concept expressed in Section 2, the External Review Group has a number of comments at various levels of detail on TILA-99, the most important of which are summarised below. These have implications for the safety assessment work and research that will need to be conducted following an approval of the Decision in Principle.

Canister failure scenarios

The canister failure scenarios considered in TILA-99 were of the "what if…" type in which a small defect was postulated or the canister is assumed to disappear completely. In the future, the External Review Group believes that this bounding approach should be augmented by a more detailed scientific approach, both within the safety assessment and in terms of evaluating scenarios based on natural processes that could potentially lead to a loss of integrity of the canister/buffer system. Examples where a more detailed evaluation is called for are:

- Evolution of pinhole defects into larger discontinuities, for example due to the expansion of corrosion products of the inner cast iron insert;
- Estimation of manufacturing defects in canisters, especially associated with the welding or closure of the outer copper vessel, including frequency, dimensions and variability;
- Common mode failure of a number of canisters, for example due to fracture movement as a consequence of post-glacial earthquakes [La Pointe & Wallmann, 1997];
- Multiple canister analysis, taking account of the range of flow parameters arising from evaluation of spatial variability in rock hydraulic properties (see later section);
- Corrosion of copper including welded seals, taking account of: (i) potential complexants including those in materials introduced into the repository, (ii) mixed corrosion product formation e.g. Fe-U-Cu, (iii) the local gas regime after perforation, (iv) salinity.

Buffer failure scenarios

The buffer is a key barrier for both assurance of longevity of copper canisters and in controlling subsequent radionuclide release to the far field. As such, it is an important barrier to evaluate.

TILA-99 examines the performance of the buffer largely through a mass transfer coefficient approach, using a lumped term Q_F that is a function of the Engineered Barrier System (EBS) dimensions, fracture spacing and flow rates, and effective diffusion coefficient of the buffer (which depends on salinity). Transparency is somewhat lost by using this lumped approach and apparently using single discrete values for the derived Q_F for each candidate site. Clearly the data on flow rate and fracture spacing at any one site show significant spatial variability that calls into question the use of single values. The same may be true with respect to salinity variations. It is understood that the motivation for the TILA-99 approach is to help discriminate among candidate sites. However, the impact of spatial heterogeneity at a given site is somewhat masked by selection of single parameter values to represent a wide range of interacting processes.

With respect to "what if…?" scenarios for the buffer, future consideration should address:

- Alteration and/or disruption of the bentonite buffer, due to chemical, thermal and mechanical effects, leading to enhanced corrosion of the

canister as well as greater release rates of radionuclides from the buffer to the host rock;
- Displacement and/or disruption of buffer by the potential expansion of the massive inner cast-iron insert of the canister;
- Relative movement of a canister and the surrounding buffer, for example canister sinking due to viscous flow of bentonite;

Rock mechanics

It is noted that the relatively high horizontal stresses and relatively low rock strength at Olkiluoto and Hästholmen could potentially lead to stress concentrations, which could in turn cause rock failure near the intersections of tunnels and deposition holes. If so, then alternative design concepts such as in-tunnel emplacement within elliptical cross-section tunnels may need to be considered and their performance evaluated in detail.

In addition, the Group makes the following recommendations with respect to future rock mechanical work:
- A technical audit should be carried out for a structured list of information requirements, in order to establish if the information is adequate for the intended purpose;
- A Finnish rock mass classification scheme should be produced to establish during excavation whether the rock mass ‘quality’ is acceptable for the disposal objective;
- A study of the parameters required for modelling the following phenomena should be carried out: i) overall rock response to excavation, ii) coupled rock mechanics and hydrogeology, iii) thermo-hydro-mechanical (THM) interactions and performance assessment;
- The Excavation Disturbed Zone (EDZ) should be studied on a site-specific basis for both rock engineering construction and as a potential fast transport pathway;
- Estimates of the time dependence of repository stability should be made, given that the design life of a repository is far longer than the usual 120 years for civil engineering structures. Also, laboratory measurements of parameters are not sufficient for repository scale predictions, as rock properties are scale dependent;
- There is no evidence that standardised methods for gathering and processing information have been used, but such protocols are required for transparency, traceability and credibility.

The significant differences in the values of the principal stress components might also impact upon postulated excavation, construction and emplacement operations within a repository, and these impacts will need to be addressed at the appropriate time.

Avoidance strategies

The Posiva application for the DIP [Posiva, 1999] mentions *acceptability criteria* for each disposal hole in the tunnel floors as well as for the layout of the whole repository. Two types of avoidance strategies are used as arguments in TILA-99, related to the identification and avoidance of negative features during the construction of the repository and selection of deposition holes. Firstly, it is assumed that ‘fracture zones’ will be avoided during the positioning of tunnels and deposition holes. This is problematic at present because of the lack of clarity in defining what is meant by ‘fracture zone’. This argument may, however, be reinstated if future research efforts make ‘fracture zones’ better defined and practically identifiable, which would make avoidance plausible. Secondly, it is assumed in TILA-99 that highly transmissive fractures will be avoided in the siting of deposition holes. This is possible on a practical basis, but is not admissible as part of the safety case because it cannot be assumed that the present hydraulic nature of a fracture will be retained over long periods of time (e.g. through an Ice Age and beyond). With the possible uncertainties over design resulting from the in-situ stresses and rock strength mentioned above, avoidance strategies are likely to become an increasingly important issue.

This issue needs to be explored in more depth, particularly as stringent criteria may result in an increase in requisite rock volume for the repository that could be more difficult to achieve.

Spatial variability

The extensive hydrogeological data at all sites reveal a high degree of spatial variability. It is not clear that the full extent of this can be fully incorporated within the concepts underlying the discrete fracture flow modelling. Moreover, the major impact of spatial variability is on transport rather than flow, but TILA-99 does not attempt to quantify this. Instead, a uniform fracture approximation is used with transport parameters that are claimed to be conservative. However, this claim is questionable in view of the upper cut-off limits that were used for selecting hydraulic conductivities and transmissivities [Hautojärvi et. al., 1995]. Other forms of data treatment (extrapolation, curve fitting) would also seem to have introduced a degree of bias. This touches on the general problem of whether it is reasonable to represent such a wide range of values (several orders of magnitude) with a single representative value for modelling purposes.

Geochemistry

High quality geochemical data have been obtained from the sites but it has not been used to its full potential in the TILA-99 safety assessment. For example, the palaeohydrogeological interpretations of past climate states discussed in underlying research reports do not appear to have been considered. Confidence in the assessment would be increased by taking into account geochemical interactions and the evolving boundary conditions resulting from climate change. At present, no justification or mechanistic basis is given for the scenarios chosen for analysis (e.g. 'disappearing canister'). The rationale behind these assumptions requires further explanation.

Better-integrated data on mechanisms affecting near-field geochemistry are required. These should address the effects of adjunct materials (e.g. cement, metals, and organics) and the possibility of interaction of waste streams given the presence of the low and intermediate level waste repository at Olkiluoto.

Most of the shortcomings in the safety analysis stem from the treatment of geochemical and hydrogeological information. Over very long timescales Posiva believe the repository will resemble a rich uranium deposit and that its impact is likely to be similar to natural U occurrences. However, this is not the concept modelled in the TILA-99 safety analysis where widespread dispersion is assumed to the extent that the repository would be 'virtually indistinguishable' from its surroundings. Recognition that the repository zone will always be geochemically anomalous would be the first step in constructing a more realistic safety case. This should encompass calculations based on the known geochemical behaviour of each radioelement in keeping with the nuclide-specific release constraints imposed by the regulators.

There is also an absence of a methodology for selecting and checking thermodynamic data used in the assignment of solubility values [Vuorinen et. al., 1998], and this has resulted in some inconsistency between the safety assessment and the sources cited. A more systematic approach to selection, evaluation and documentation of chemical data is suggested for the future.

Information on rock matrix diffusion at the four sites is lacking. This raises questions concerning the use of a 'transport resistance' concept including parameters that have not been measured and are difficult to estimate.

It is noted that the 'stacked water model', where waters of different ages are assumed to be flushed downwards through the system, suggests considerable rock mass permeability. It is not clear whether this geochemical perspective is consistent with the relatively low measured permeabilities.

Finally, the hydrochemistry of the rock matrix has not been well characterised. This has the potential to increase the salinity of the near-field pore water, and should be studied further.

Microbiology

A more integrated, holistic view should be taken of the potential for microbiological action during operational and post-closure phases. Assessment should include the impact of constructional materials, a potentially prolonged oxidative phase, and impacts arising from gas generation and of gradients in pH, Eh, etc., as well as the potential for nutrient transport.

Traceability

As noted in Section 2, the reports are generally well presented. Nevertheless, there are some problems of data traceability, which make it more difficult for the reader to assess some of the conclusions. This is particularly the case for the development of the structural models from the site-specific structural data. Also, it is difficult to see the interface between the bentonite-to-rock transfer coefficient values (Q_F) and flowpath transport resistance values (WL/Q) presented in TILA-99 and the underlying information on these parameters in the site scale and site-to-canister scale reports. Consequently the choice of ranges in these crucial values used in the safety assessment is not as clear as it could be.

Biosphere dilution

The degree of biosphere dilution assumed in TILA-99 does not appear to be conservative as claimed. For example, it is an order of magnitude larger than the 10^4 m^3 per year assumed in SITE-94 [SKI, 1997] and the drinking water pathway is not necessarily the most conservative [BIOMOVS, 1996]. Potential well dilution factors will depend on the type of well considered and on whether a single or multiple canister analysis is being carried out. Different types of well may exist on the site at different times in the future. It is to be hoped that more sophisticated biosphere analyses will be featured in the next round of Posiva performance assessment development which address these issues, among others. For example, it should be acknowledged that biosphere processes also have the potential to concentrate radionuclides.

Scenario selection and analysis

TILA-99 presents a good descriptive evaluation of the future evolution of the natural environment of a repository at each site, which outlines the main potential impacts of glacial cycling over the next hundred thousand years. However, it is rather disappointing that the quantitative analysis and computational scenarios do not build on this convincingly.

The cases analysed are not linked to the evolutionary descriptions and thus it is not possible for the reader to evaluate their real significance, and important correlations can be lost. For example, the ‘noben’ scenario (i.e. what if very poor bentonite) concentrated on the effect of modifying the mixing volume for released radionuclides, rather than on the critical issue of the effect of a degraded diffusive barrier on canister lifetimes and the diffusive constraint on radionuclide release rates.

A further example is the gas scenario, which only looks at gas expulsion of radionuclides from a canister, rather than asking the question about what is actually likely to happen physically in a canister and the buffer as gas is produced and migrates. The issues ‘missed’ are the generation of steel corrosion products (the source of the gas) which may expand and increase the size of the aperture in the canister, as well as displace and disrupt the encompassing buffer layer. Thus a more integrated approach to gas generation needs to be taken in the future.

The approach to scenario development in TILA-99 is not commensurate with the current state-of-the-art, and would not be suitable for the PSAR due in year 2010, which needs to demonstrate compliance with the relevant regulations [Ministry of Trade and Industry, 1999]. For the PSAR there needs to be a more systematic and comprehensive approach to scenario development, which considers how various process system Features Events and Processes (FEPs) and External FEPs (EFEPs) interact to affect the performance of the proposed repository.

Introduced materials and other wastes

The longer that repository excavations are left open and ventilated, the more likely exchange can occur in volatile components, such as H_2O, O_2, N_2, CO_2, CH_4, H_2S. Exchange of such volatiles can, in turn, promote various and possibly irreversible mineral reactions. Together, these changes could degrade the ability of the rock to effectively buffer chemical conditions in the future. Possible future consideration should be devoted to such scenarios to examine if there are any significant impacts on repository performance.

Various materials are likely to be introduced into a repository during construction, for example rock anchoring systems, trunking for ventilation, transport fuel and other infrastructure. While much of this could be stripped out before closure, some materials could remain and alter the Eh, pH and complexant concentrations. For example, any stray organic materials could lead to enhanced microbiological activity.

A related problem is the disposal of reactor operating and decommissioning wastes in proximity to, and as part of a spent fuel repository. These and any remaining concrete seals could again change the groundwater composition, for example by creating a high pH plume, and adversely affect the near-field barriers. At an appropriate stage in the repository programme, these issues should be assessed in detail.

Conservatism and key safety factors

There is a formal requirement for conservatism in presenting a safety case expressed in Statute 478 [Ministry of Trade and Industry, 1999] and this challenge has been adopted vigorously in TILA-99. Whilst this is clearly a sensible approach for a programme to take, in that safety is apparently better assured, *a priori* system-wide conservatism in selection of models and data can ultimately lead to a smoothing or blurring of quantitative results and a consequent lack of transparency in the understanding and presentation of realistic system and sub-system behaviour. It can thus result in the opposite of what is intended by the requirements. This is seen to some extent in TILA-99, where it is difficult to identify clearly those components of system performance that contribute most to long-term safety. From the reviews carried out by the External Review Group, it appears that, once a canister is assumed to have failed, the key safety factors are: size of the hole in the canister, release rates, solubilities, the transfer coefficient from bentonite to rock (Q_F), and well dilution.

Comparison with other assessments

It would have been valuable were TILA-99 to have compared its approach, assumptions and results with those of related assessments produced in other countries [NEA, 1997]. This would have illustrated better the stated conservatisms of this assessment and highlighted any significant differences in approach. Also, it would have been useful to have made a comparison with the safety assessments for the existing shallow geological disposal facilities at Olkiluoto and Hästholmen.

Retrievability

Statute 478 contains a requirement that wastes should be retrievable and the Posiva application for the Decision in Principle [Posiva, 1999] says that retrieval can be implemented using available technology. However, this is not discussed explicitly in TILA-99 (although more recent work has looked at this issue). Eventually, it would need to be demonstrated in practice that this is indeed the case, and that no design modifications that might affect the design basis safety concept would be needed in order to facilitate retrieval.

Overall implication of comments

Although the above comments may appear extensive, this is largely a reflection of the breadth of information and analysis required by an assessment such as TILA-99, and of the need to ensure focus, as a future programme goes site-specific. None of the comments challenges the fundamental safety concept proposed by Posiva, or provides grounds for refusing the Decision in Principle. However, the comments do have implications for the future Posiva and STUK programmes, and these are discussed in Sections 5 and 6.

4 OLKILUOTO

The External Review Group has given some consideration as to whether Olkiluoto is the best of the four sites and whether it is a workable site. Differences in geochemical evolution excepted, it is difficult to distinguish between sites based solely on geological or hydrogeological characteristics because (i) the characteristics of the four sites are closely similar, especially if spatial variability within sites is considered, and (ii) the low-permeability buffer effectively mitigates the coupling between the performance of the engineered barrier system (EBS) from the hydrogeology of a site. It also is noted that there are no clear criteria for preferring saline or non-saline sites, and that glacial cycling between saline and fresh water at repository depths at the coastal sites leads to further complexities. However, the costs and non-nuclear environmental impacts of transporting spent fuel clearly favour the coastal sites of Olkiluoto and Hästholmen.

An important selection factor is whether a site is large enough to accommodate all the projected waste. In this context, it is not obvious from the information provided whether wastes from additional reactors can be accommodated into the geological structure at Olkiluoto or the other sites, without some difficulty regarding 'respect distances' from important rock discontinuities. This is an issue which needs to be explained and reviewed more fully in the near future, taking into consideration the following points:

- Section 14.4 of TILA-99 outlines some of the layout problems that would be encountered at each of the sites as a result of geological structure, rock stresses and the variation of rock strengths. However, there is no real discussion of the practical or performance assessment implications of this. It would be useful to know whether Posiva considers that any aspect of the safety analysis would be sensitive to a larger or more dispersed repository.
- If there is a real constraint on useful rock volumes (particularly at Olkiluoto) then what is the next option for Posiva if either the existing waste projections or the extended projections cannot be accommodated? Would it be a matter of building a second repository in the same general area (i.e. within a few kilometres) or would some other solution need to be found?
- The rate of waste arisings over the next few decades would also need to be taken into account. For the extended waste projections from new reactors, it may be that a decision on how to handle the additional waste volumes (if they pose a practical problem at Olkiluoto) could be postponed until many years into the future. The implications of this need to be part of the present review process.
- The implications of conceptual model uncertainty arising from alternative geological structure interpretations and models of groundwater flow at Olkiluoto should be assessed.

On the basis of present evidence, Olkiluoto is thought likely to be a workable site, with the following positive and negative attributes:

Positive attributes

- Reducing chemical conditions;
- Waste accommodated in a single repository panel;
- Sea-borne transport of wastes to the site;
- Local acceptance;
- Large biosphere dilution for any released radionuclides due to discharge to seabed for current inter-glacial period;

- Low flow of saline water during current interglacial period;
- Potential hydraulic shielding by sub-horizontal zones;
- Evidence from VLJ repository suggests it is a relatively low permeability site.

Negative attributes

- Relatively high horizontal rock stresses;
- Relatively low and anisotropic rock strength;
- Potential impact of saline water on backfill and buffer;
- Uncertain hydrogeological evolution;
- Most complex hydrochemistry.

The Posiva programme has now reached the point when it is not possible to learn a great deal more without going underground and exploring the real volume of rock in which a repository might be located. If work goes ahead at Olkiluoto then it will provide the opportunity to establish the variation in rock stresses and rock strengths, and refine conceptual models of rock structure, groundwater flow and of palaeohydrogeological evolution of the site, that will increase confidence in calculations designed to scope its likely future behaviour. Posiva should be encouraged to define how their programme of work at the site will achieve both these, and other, design-related objectives, before the programme advances much further.

Posiva has suggested Hästholmen as a second choice should Olkiluoto prove impractical. It is not apparent that the choice can readily be based on the geological evidence, although it is appreciated that non-geological factors must also be given high weighting. It is also not clear why only the volume of rock beneath Hästholmen is considered, rather than a broader area that may provide more flexibility in repository location.

5 IMPLICATIONS FOR POSIVA

It can be seen from the preceding discussion that the External Review Group is in favour of approving a Decision in Principle for Posiva to continue with its repository development programme at Olkiluoto. However, there are several areas where the Group feels that this approval ought to be conditioned by specific requirements for Posiva to carry out more research and development. Some of this work is likely already to be in Posiva's forward plans, but it is important, nevertheless, for STUK to identify these issues clearly.

The External Review Group divided these issues into three categories:

Category 1: Issues that should be clarified or resolved in the near future (i.e. before locating and driving the shaft and/or access tunnel) using existing data;

Category 2: Issues that should be included as points of focus within the Posiva R&D programme and be resolved by 2010;

Category 3: Issues which Posiva should devote resources to in further site characterisation work at the surface or in the underground rock characterisation facility at Olkiluoto.

It should be noted that some of the Category 3 issues also require addressing in the immediate future. The issues identified within each of these three categories are outlined below.

Category 1

Further explanation is required on the issue of avoidance strategies (avoidance of adverse rock conditions or geological structures) which would allow a repository to be fitted into the rock mass of the Olkiluoto site. This needs to address both the basic structural geology of the site and the potential large variations in size of the repository, dependent on the amount of fuel in the eventual disposal inventory. To this end, it would be useful if the structural data from the site could be reprocesses and interpreted by at least two different and independent groups, and their implications identified and quantified. This would also provide an improved background for the development of the Finnish rock mass classification scheme mentioned earlier.

As discussed above, it is not clear that the rock stresses at potential repository depth, combined with the available data on rock strength, would allow flexibility in locating and constructing a repository at Olkiluoto without the use of significant support work or design modifications (e.g. in-tunnel as opposed to KBS-3 type emplacement). The implications of this on repository design and performance need to be evaluated.

Owing to the difficulty in verifying the degree of conservatism in TILA-99, it is necessary for Posiva to explain more clearly which factors in system performance are, realistically, most sensitive and, hence, which features are the principle supports of the safety concept. A dialogue with STUK could be initiated to discuss the issue of conservatism and its implications for deriving the specified performance measures.

Category 2

A more sophisticated performance assessment methodology should be developed and tested before the 2010 PSAR is submitted. This needs to include treatment of the following matters:

- A more realistic approach to biosphere modelling which follows current international developments;
- More traceable data derivation, presentation and management, in particular fuller explanation of the component factors within lumped performance terms, such as mass-transfer coefficients;

- Improvement in the calculational capabilities (e.g. explicit treatment of radionuclide decay chains during transport) and flexibility of performance assessment codes to approach the current state-of-the-art;
- Systematic and comprehensive treatment of alternative scenarios that gives particular attention to possible failures or uncertainties in key components of the repository that dominate the safety case (e.g. intact buffer, long-lived canister, constant-sized defects, stable mechanical and chemical conditions of the site);
- Evaluation of buffer-degradation scenarios, including possible expansion of corrosion products of the cast-iron insert and gas generation within a waste container, on the performance of the Engineered Barrier System (EBS). Similarly, the effects of gas pressure on seals needs to be evaluated;
- The impacts of time dependent, repeated or cyclical processes need further consideration, particularly in the period between ten thousand and some hundreds of thousands of years to account for glacial cycle effects;
- A more detailed analysis of canister degradation during the thermal period, particularly with respect to impurities in the buffer which may contribute corrodants;
- A fuller analysis of the various failure modes of the canister, moving away from the constant pinhole model to examine the potential for time-evolution of the size of this initial manufacturing defect (for example related to iron corrosion), and the use of multiple canister analyses;
- The impacts of a requirement for waste retrievability on the design and performance of the system;
- The impacts of materials necessarily introduced into the repository during construction: grouts, cement and concrete/steel structures.

Further consideration should be given to the impacts of saline waters (and cycling fresh and saline waters) on the performance of the EBS and on the glaciation-driven dynamics of the groundwater system located in a present-day coastal site. The hydrogeochemistry interpretation and predictions suggest that glacial and post-glacial events can introduce a wide divergence of far-field/near-field geochemical conditions.

Design alternatives for the EBS need to be explored and related to growing knowledge of the Olkiluoto site. In particular, the in-tunnel, elliptical cross-section, versus the TILA-99 in-borehole design may need to be assessed in detail.

Category 3

Posiva should develop an Olkiluoto site characterisation strategy and an underground experimental strategy soon, before the siting of a shaft. These strategies need to include consideration of the following issues:

- Defensible establishment of baseline hydrogeological and hydrochemical conditions before construction begins to perturb the site;
- Better characterisation of the hydrochemistry of the rock matrix and the dynamics of the fresh and saline waters in the system;
- Development and testing of alternative conceptual models of the geological structure and groundwater flow regime at the site;
- Characterisation of spatial variability in rock mass hydraulic properties;
- In-situ rock stress measurements and tests/demonstrations of alternative construction and waste emplacement techniques, with emphasis on the rock response to excavation.

6 IMPLICATIONS FOR STUK

The External Review Group believes that there should be a close interaction between STUK and Posiva over the coming years, with well-identified requirements and regular milestones to assist the regulatory process. In addition, TILA-99 spells out a number of challenges that STUK will need to address in order to provide proper guidance to Posiva. We deal with these first.

TILA-99, either directly, or by implication, can be said to have brought into focus three main areas where some regulatory consideration is needed, well before any formal licensing submission is made. The first of these areas concerns the formal requirement for conservatism in presenting a safety case, expressed in Statute 478 [Ministry of Trade and Industry, 1999]. Whilst this is clearly a sensible approach, it can lead to a lack of transparency, and may have an undesirable impact on the Posiva assessment methodology in that a proper description of expected system development may not emerge clearly, and parameters and values may be chosen which are acknowledged to be unrealistic. It is not reasonable to specify a defensible conservative case *a priori* before the system has been adequately characterised and potential interacting effects have been considered. It can thus result in the opposite of what is intended by the requirements. STUK may wish to give some thought and advice on how and at what stage a reasonable degree of conservatism can be incorporated in future assessments. Ideally, an attempt would first be made to represent possible evolutions of the disposal system in order to gain an understanding of critical issues, and then a margin of safety could be introduced.

Secondly, further consideration needs to be given to the most appropriate safety indicators at times beyond about ten thousand years into the future. The averaging of release concentrations over long periods may not be the most appropriate way of evaluating the impacts of future environmental changes that occur with much shorter periodicity.

The third area concerns the treatment of human intrusion into a repository. It is widely acknowledged (as noted in TILA-99) that it is inappropriate to compare possible radiological impacts of intrusion to standards designed to regulate for the undisturbed evolution of the system. A regulatory position needs to be adopted on how to weigh the significance of intrusion impacts when making decisions about aspects of repository acceptability. Such a position needs to be developed within an international framework, and in a broader environmental and social context of decision making than that simply of radioactive waste disposal, and STUK may wish to consider how to approach this.

Each of these matters might affect the future direction of STUK's own R&D programme to develop its regulatory capabilities.

STUK may wish to keep closely abreast of the Posiva programme so that it can act expeditiously when called upon to express a regulatory view. This could also help to ensure that, when formally called upon to do so in several years time, surprise issues are less likely to emerge and cause problems on all sides. In order to accomplish this, STUK may wish to consider establishing a regular review process with Posiva. Components of such a formal process might include:

- Written clarification of (for example) Category 1 issues;
- Review of Posiva’s site characterisation programme forward plans (before shaft construction begins);
- Review of Posiva’s underground experimental programme forward plans (before shaft construction begins);
- Regular (e.g. every three to four years) review of the content and achievements of Posiva’s R&D programme (equivalent to the reviews of

SKB's RD&D programme [SKB, 1998a,b]);

- Review of interim safety analyses, prior to PSAR, to track the development of the Posiva performance assessment methodology;
- Implementation of key elements of an independent assessment capability (e.g. scenario development) to increase public confidence in the decision for repository construction.

The objective of each of these items would be to ensure that STUK's own regulatory concerns are being taken into account by Posiva as their programme develops. This approach would clearly place additional demands on Posiva in terms of production of documents that would need to be considered in the context of the national disposal programme.

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