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Strategic Action Plan for Implementation of European Regional Repositories: Stage 2

Work Package 4

Safety and Security of Regional Repositories

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Summary

The basic concept within both SAPIERR I and SAPIERR II is that of one or more geological repositories developed in collaboration by two or more European countries to accept spent nuclear fuel, vitrified high-level waste and other long-lived radioactive waste from those countries.

The SAPIERR II project (Strategic Action Plan for Implementation of Regional European Repositories) examines in detail issues that directly influence the practicability and acceptability of such facilities. To achieve this it is necessary this is to consider the complete chain of activities and facilities that would be needed take radioactive waste from storage facilities at nuclear power plants, or from centralised national storage facilities, to final disposal in one or more shared deep geological repositories.

This report is produced under Work Package 4 of SAPIERR II, the aim of which is to make an outline examination of the safety and security aspects of implementing one or two regional repositories within the European Union, relative to a larger number of national repositories. The focus is on nuclear safety (i.e. radiological safety) and nuclear security.

The emphasis in this report is:

- to survey the safety and security standards that would apply to a multi-national radioactive waste management system leading to final disposal within one or more shared repositories in the EU;
- to confirm that methods and techniques are available to assure safe and secure accomplishment of all the necessary waste management steps, and to indicate their performance;
- to make simple generic comparisons and assessments of safety and security aspects of implementing such a system, compared to that of implementing a number of national systems.

High levels of safety and security will be applied to the management and final disposal of radioactive waste and spent nuclear fuel in both national and shared projects.

This report shows that the required safety and security standards are achievable for all required steps and confirms that a shared project presents no technical issues that will not have to be overcome in national projects. International treaties and conventions, and the comprehensive system of international guidance, national regulations and control mechanisms, ensure that a shared regional repository and associated waste management system will be at least as safe and secure as any national repository and waste management system.

Are there safety benefits in developing multinational repositories?

The assessed radiological safety of notional shared waste management systems shows a small collective dose reduction relative to national waste management systems of the same capacity. This arises from an assumption that timely development of a shared repository would reduce the average time that spent fuel is stored at national facilities. The dose saving related to shorter storage times outweighs the small increase in collective doses due to longer transport distances. The calculated net collective dose reductions (to workers and to members of the public) are small however, only about 1/1000th of the collective doses from the reactor operations that produced the waste.

A shared waste management system and final repository offers a potential safety advantage over separate smaller national systems primarily as a result of the pooled financial and human resources that can be invested to ensure implementation to high technical standards. The shared responsibility and multinational oversight should also give greater assurance of regulatory control and adherence to the strict international safety criteria and requirements.

Are there security benefits in developing multinational repositories?

A qualitative assessment of the physical protection of a shared waste management system and final repository relative to national waste management systems indicates the security risks are similar, and in both cases less than the security risk posed by operating nuclear reactors. This arises primarily because operating nuclear reactors represent a more sensitive target from economic and safety perspectives. The increased number of international shipments, and longer transport distances, introduce an increased risk of attacks against spent nuclear fuel in transit, but the nature of the material and robust containment systems mean that even a successful attack could not produce serious radiological impacts.

A shared waste management system offers a potential security advantage as a result of the pooled protection and intelligence resources that can be applied to ensure physical protection. A shared final repository also offers a security advantage in the long-term against proliferation of nuclear materials, since the number of sites at which nuclear material is located is reduced.

Co-operation and timely implementation

For both security and radiological safety, a general benefit of the development of a shared waste management system is that a well-focussed, co-operative effort from several countries can lead to a fuller and more critical consideration of safety, security and other issues at each step, and thus a better quality of implementation may be achieved.

We also consider that the combined efforts of several countries may give better prospects for joint realisation of a project at an earlier time than if national projects proceed independently. This presents a small but tangible benefit due to a reduction in the average time that spent fuel is stored at national facilities, and also a less quantifiable benefit of less chance that disposal will be indefinitely delayed in any country. We fully support, however, the view of the IAEA group on developing multinational radioactive waste repositories that:

> *"the improvements in safety and security that are expected are at a global scale. It is not intended to imply that a multinational repository will be safer or more secure than a properly implemented national repository. The global benefit results from making a proper disposal facility accessible also to countries that may not be in a position to implement a state of the art national repository."*

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1 Introduction

1.1 Background

Soon after the peaceful use of nuclear energy began to develop in the 1960s and 70s there were proposals for multinational solutions to provide fuel cycle services to power plant operators [1]. For the final steps in the cycle, the management and disposal of spent fuel or radioactive waste, it was only reprocessing services that were implemented multinationally; these were provided by countries such as France, the UK and Russia[1].

Interest in multinational disposal revived in the late 1990s, driven by the high costs of geological repository programmes and also by the security concerns associated with the prospect of fissile material being widely distributed across the world. Although several initiatives were proposed, none led to success, partly because the proposed approaches were judged to be premature and too commercial. Accordingly, in 2002, the not-for-profit organisation, Arius (Association for Regional and International Underground Storage), was established to help partner organisations from various countries explore the possibilities of shared disposal facilities. The current growing interest in initiating or expanding nuclear power programmes also emphasises the need for all countries to have a credible long-term management strategy for high-level and long-lived radioactive waste. For many, especially new or small programmes, multinational cooperation leading to shared disposal facilities could be an attractive option.

In Europe, the Parliament and the EC have both expressed support for concepts that could lead to regional shared facilities being implemented in the EU. The EC has funded two projects that can form the first steps of a staged process towards the implementation of shared regional or international storage and disposal facilities. In the period 2003 to 2005, the EC funded the project SAPIERR (Support Action: Pilot Initiative for European Regional Repositories), a project devoted to pilot studies on the feasibility of shared regional storage facilities and geological repositories for use by European countries. The SAPIERR I project looked at the basic technical and economic feasibility of implementing regional, multinational geological repositories in Europe. The studies [2; 3; 4] indicated that shared regional repositories are feasible and that a first step could be to establish a structured framework for the future work on regional repositories.

1.2 The SAPIERR II project

The SAPIERR II project (Strategic Action Plan for Implementation of Regional European Repositories) examines in more detail specific issues that directly influence the practicability and acceptability of such facilities. If these are to become a reality, a dedicated organisation will be required that can work towards the goal on the extended timescales that national disposal programmes have shown to be necessary. Specific terminology is introduced in the SAPIERR II project to describe the organisations that may be formed to perform the work leading to implementation of a shared repository in Europe. The terms are as follows:

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1 These countries originally also provided a disposal service since they did not return reprocessing waste to their customers. With time, however, a waste return clause was included in new reprocessing contracts – mainly as a reaction to public and political pressures in the reprocessing countries.

- European Development Organisation (EDO): the initiating, non-profit organisation for a shared geological disposal facilities project. Its objective is to establish the systems, structures and agreements and carry out the work needed to put in place a shared waste management solution and geological repository (or repositories). This work would continue through the investigation of potential sites and up to the point of license application to begin the construction of a repository. It is assumed that this will take about 10 or more years. At this point the EDO may decide to transform into or separately establish the ERO.

- European Repository Organisation (ERO): the implementing organisation for waste disposal. The ERO would be the license holder for the repository and responsible for all subsequent operational activities in a host country that has agreed to dispose of waste from other European countries. The form for the ERO will be chosen at a future date by the members of the EDO, assuming that they come to the conclusion that the EDO organisation needs to be altered. The choice will also be strongly influenced by the preferences of the country or countries that have been identified as repository hosts. The ERO could be either non-profit or commercial in structure.

The goal of SAPIERR II (2006-2008) is to develop possible practical implementation strategies and organisational structures that will enable a formalised, structured European Development Organisation (EDO) to be established after 2008 for working on shared EU radioactive waste storage and disposal activities. The tasks in the SAPIERR II project are listed below. Each task translates into a Work Package (WP), as follows:

1. Preparation of a management study on the legal and business options for establishing a European Development Organisation (EDO).

2. A study on the legal liability issues of international waste transfer within Europe.

3. A study of the potential economic implications of European regional storage facilities and repositories.

4. Outline examination of the safety and security impacts of implementing one or two regional repositories relative to a larger number of national facilities.

5. A review of public and political attitudes in Europe towards the concept of shared regional repositories.

6. Development of a Strategy and a Project Plan for the work of the EDO.

7. Management and dissemination of information.

The basic concept within both SAPIERR I and SAPIERR II is that of one or more geological repositories developed in collaboration by two or more European countries to accept high-level and long-lived radioactive waste from those countries. In SAPIERR and other international discussions, this has been termed the “regional” repository concept, although the term “shared” repository is also appropriate. In this report, the terms regional and shared repository (or waste management system) are used interchangeably. It is to be understood, however, that neither term imply a specific regionalisation of proposals or specific arrangements for sharing, which would be a matter for any countries or national organisations that decided to consider participating in a future EDO as defined above.

The SAPIERR II project does observe the boundary condition that the countries involved are assumed to be members of the European Union and that the repository or repositories referred to will be sited within the territory on one or more of those EU countries.

1.3 Objectives and scope of this report (safety and security)

This report is produced under Work Package 4 of SAPIERR II, the aim of which is to make an outline examination of the safety and security aspects of implementing one or two regional repositories within the European Union (EU), relative to a larger number of national repositories.

A basic premise for this task, and indeed for the SAPIERR II project, is that it is technically feasible to develop storage facilities and final repositories for radioactive waste that are both safe and secure as judged against international standards and the relevant European Directives. This is demonstrated by the facts that:

- radioactive waste, including spent nuclear fuel, has been (and is) transported over significant distances and across national borders both within and outside the EU;
- stores for radioactive waste, including high-level waste and spent nuclear fuel, operate in many countries of the EU;
- underground repositories for short-lived, low-level and non-heat generating radioactive wastes operate in several EU countries; and
- projects for the development of deep geological repositories for high-level, heat-generating and long-lived radioactive waste are proceeding in several EU countries.

Rigorous assurances of safety and security are essential requirements for all such activities and facilities, and all such projects are developed within a framework in which safety and security have the highest priority. However, with proper attention to design, siting, quality of implementation, monitoring and control, and provided that the necessary financial and technical resources are committed, the feasibility of achieving the required standards of safety and security is not in question.

The emphasis in this report, therefore, is:

- to survey the safety and security standards that would apply to a multi-national radioactive waste management system leading to final disposal within one or more shared repositories in the EU;
- to confirm that methods and techniques are available to assure safe and secure accomplishment of all the necessary waste management steps, and to indicate their performance;
- to make simple generic comparisons and assessments of safety and security aspects of implementing such a system, compared to that of implementing a number of national systems.

The radioactive wastes considered are the same as considered in SAPIERR I [5], that is spent nuclear fuel (SF) from commercial power plants, high-level waste (HLW) from the reprocessing of spent fuel and long-lived low and intermediate-level radioactive waste (L/ILW-LL), not suited for disposal in near-surface facilities (see [6]).

The radioactive waste management system considered is the complete chain of activities and facilities that would take such radioactive wastes safely and securely waste from storage facilities at nuclear power plants, or from national interim storage facilities, to final disposal in one or two regional deep geological repositories.

In this report:

Chapter 2 discusses aspects of safety and security and defines the scope of this report, which is focused on nuclear safety and security. It also outlines the general approach to achieving safety and security, and indicates a development strategy in which safety and security requirements are integrated into the decision making for a project.

Chapter 3 outlines the boundary conditions that must be met by any national or multinational system for the long-term management of radioactive waste in the EU. This includes obligations under international treaties and agreements and the internationally-developed objectives and principles related to safety, security and nuclear safeguards. Societal acceptance, political decisions and costs will also be important boundary conditions, but are not discussed in this report.

Chapter 4 sets out a brief description of the options and scenarios for a shared regional disposal system consisting of one or more deep geological repositories, encapsulation facilities, and transport and transfer arrangements. This includes consideration of inventory, number of repositories, host rock options, transport distances, timing and operational steps. The options and scenarios are based on those developed in SAPIERR I and consistent with those considered in the SAPIERR II report on economic aspects.

Chapter 5 discusses nuclear and radiological safety aspects of radioactive waste management from waste acceptance to disposal. This includes discussion of safety standards, waste and waste package acceptance, transport safety, operational safety of facilities, and repository post-closure safety. Proportionately, most attention is directed at safety related to spent nuclear fuel, which is radiologically the dominant waste form. The final section of the chapter provides a safety overview and presents indicative estimates of radiological impacts for a shared waste management system and equivalent capacity national systems.

Chapter 6 discusses security aspects of a radioactive waste management system from waste acceptance to disposal. This includes discussion of nuclear security standards, defining and countering security threats, and physical protection systems in general terms. Security aspects of a shared waste management system and its stages are then discussed, and conclusions are drawn on the security of a shared system compared to a case of several smaller national systems.

Chapter 6 discusses physical protection aspects of security of a radioactive waste management system from waste acceptance to disposal. This includes discussion of nuclear security standards, defining and countering security threats, and physical protection systems in general terms. Physical protection of a shared waste management system and its stages are then discussed, and conclusions are drawn on the security of a shared system compared to a case of several smaller national systems. The chapter mentions non-proliferation and nuclear safeguards aspects, but these are not discussed in detail.

Chapter 7 presents a summary of the conclusions that are supported and discussed in the preceding chapters, plus final remarks on common factors that underpin safety and security.

2 Safety and security strategy

This Chapter discusses aspects of safety and security and defines the scope of this report, which is focused on nuclear safety and security. It also outlines the general approach to achieving safety and security, and indicates a development strategy in which safety and security requirements are integrated into the decision making for a project.

2.1 Aspects of safety and security

Safety and security are broad terms and could, at their most general, include consideration of many issues, e.g. road safety, fire safety, personal security, security of resources. For a radioactive waste management system, such as considered here, the relevant issues can be broadly classified as conventional safety and security, and nuclear safety and security, as indicated in Box 2.1.

Box 2.1: Aspects of safety and security for a radioactive waste management system

Conventional safety

The protection of workers and the public.

Prevention, minimisation or mitigation of consequences of:

- transport accidents (waste and construction traffic);
- construction/mining accidents (above and underground facilities);
- operating accidents (machinery use, maintenance, dropped loads etc.).

Minimisation of and protection from toxicological hazards.

Nuclear safety

The radiological protection of workers and the public.

Prevention, minimisation or mitigation of conventional accidents (see above) that may have radiological consequences.

Control of radiation sources, including containment and shielding of the radioactive waste, and criticality assessment.

Assessment and monitoring of radiation doses to workers and the public from transport and facility operations (normal operations and accidents).

Assessment of radiation doses to workers and the public after repository closure.

Conventional security

Prevention, detection and response to, theft, sabotage, unauthorized access or other malicious acts not involving the radioactive wastes, e.g. at sites in commissioning.

Nuclear security

Prevention, detection and response to, theft, sabotage, unauthorized or malicious acts involving the radioactive wastes or their associated facilities, perpetrated by terrorists or other non-state agents. This includes:

- considering damage to facilities and loss of radioactive materials, especially with potential for radiological and nuclear consequences;
- protection of facilities and radioactive waste through hardware, personnel, procedures and design.

Prevention, detection and response to removal, diversion or misuse of nuclear materials by facility operators or state agents. This means actions in contravention of international nuclear safeguards agreements, in particular, to divert nuclear materials from civilian nuclear power cycle to military uses.

It is important that all aspects of safety and security are considered and that adequate levels of protection are provided at all times, and for all steps, against all potential hazards and potential threats. The attention given to each hazard and threat may vary, however, at different stages of the development.

In this report, the focus is on nuclear safety (i.e. radiological safety) and nuclear security. This is because it is the nuclear and radiological aspects that are the special and defining aspects of the proposal for regional deep geological repositories in the EU. They are, therefore, the most important aspects to consider at this conceptual stage.

Box 2.2 gives the International Atomic Energy Agency (IAEA) definitions of safety, security and protection in the nuclear and radiological context.

Box 2.2: IAEA definitions of safety, security and protection

The IAEA Safety Glossary [7] makes the following definitions and comments.

(nuclear) safety

The achievement of proper operating conditions, prevention of accidents or mitigation of accident consequences, resulting in protection of workers, the public and the environment from undue radiation hazards.

(nuclear) security

The prevention and detection of and response to, theft, sabotage, unauthorized access, illegal transfer or other malicious acts involving nuclear material, other radioactive substances or their associated facilities.

This includes, but is not limited to, the prevention and detection of, and response to, the theft of nuclear material or other radioactive material (with or without knowledge of the nature of the material), sabotage, and other malicious acts, illicit trafficking and unauthorized transfer.

The glossary notes that the terms are often abbreviated to safety and security, respectively. It also notes:

There is not an exact distinction between the terms safety and security. In general, security is concerned with malevolent or negligent human actions that could cause or threaten harm to other humans; safety is concerned with the broader issue of harm to humans (or the environment) from radiation, whatever the cause.

protection and safety

The protection of people against exposure to ionizing radiation or radioactive materials and the safety of radiation sources, including the means for achieving this, and the means for preventing accidents and for mitigating the consequences of accidents should they occur.

The glossary notes:

Safety is primarily concerned with maintaining control over sources, whereas (radiation) protection is primarily concerned with controlling exposure to radiation and its effects. Clearly the two are closely connected: radiation protection (or radiological protection) is very much simpler if the source in question is under control, so safety necessarily contributes towards protection.

By the IAEA definitions, security might be considered as a special topic within the broader subject of safety. In this report, we keep the issues of safety and security separate and on equal footing. We note, however, that safety and security are closely connected and the strategies, design features and measures to promote one may, in many cases, be beneficial to the other. Special attention needs to be drawn to any cases where the requirements for safety and security conflict.

2.2 Achieving safety and security

Some of the hazards and threats relevant for the development of radioactive waste management system are common to many other industrial, mining and infrastructure developments. Hence, the methods for assessing and controlling such hazards and threats are well established, e.g. related to transport accidents, construction accidents, transport and handling of heavy loads, mining safety, seismic hazard, site security etc. In this case, as long as the engineering project and design remains within precedent practice, then it can be assumed that adequate levels safety and security can be achieved against such threats. Therefore, minimum attention needs to be given to these issues at early stages of concept and design development, rather, they will come into focus as actual designs and plans for implementation are developed and siting is considered.

For most hazards and threats, safety and security are achieved by a process of:

(1) safe and secure design;

(2) implementation of safety and security features, controls and procedures;

(3) monitoring of safety and security performance.

This is the case for the conventional hazards and threats mentioned above and also for the nuclear safety and security hazards during transport and facility operations. In this process, the design and implementation of safety and security features and controls can be based on precedent practice for similar facilities, but the last element of monitoring of performance is crucial. It gives confirmation that adequate levels of safety and security are being indeed being achieved or, if not, warning of the fact so that additional features can be installed, procedures enacted, or the process halted until a method of overcoming the particular problem can be implemented.

A defining principle of the geological disposal concept is that, after repository closure, the disposed waste should remain safe and secure even without monitoring or further protective actions [8]. Monitoring may be carried out, at least for a period following closure, and it is possible that actions could be taken to correct any recognised problems. It is a design principle, however, that such monitoring or remedial actions should not be relied on. This is an unusual challenge and a very demanding one, especially in the case of a repository for long-lived radioactive waste, i.e. wastes may remain hazardous for hundreds of thousands of years, e.g. see [9].

Thus, although the stages of transport and operation offer more potential for accidents and potential safety and security risks, these stages are subject to active control, supervision and possibility of correction. After repository closure, the potential for accidents and the safety and security risks will have been reduced to very low levels, but it is more challenging to prove that adequate levels of safety and security will be maintained into the future.

This report discusses achieving nuclear safety and nuclear security:

(1) during radioactive waste and nuclear material transport;

(2) during operations of facilities (be they encapsulation, storage or disposal facilities);

(3) after final closure or a disposal facility or repository.

2.3 A development strategy anchored on safety and security

Radioactive waste management comprises a series of steps, such as inventorying the waste, processing and conditioning of the waste, interim storage and final disposal. Throughout this process decisions must be taken to deal practically and efficiently with the situation in hand and with a view to the further steps that must be taken. Likewise, individual waste management facilities, and especially the repositories for final disposal of the radioactive waste, are planned and implemented in a step-by-step manner, taking account of the current situation and knowledge but heading towards a final goal, e.g. see [8].

For radioactive waste management and its final disposal, the highest-level goal and objective is safety (in all its aspects) and security, which must be assured both during all waste management operations and, in the long term, after disposal is complete.

A strategy is needed, therefore, to progressively design and implement elements of the system that are matched to the current technical, regulatory and socio-political conditions, while keeping all elements and activities in compliance with relevant standards, and keeping the goals of safety and security in strong focus. That is, the safety and security requirements should be into integrated into the step-by-step decision making for the project.

Figure 2.1 illustrates such a strategy anchored by safety and security objectives and requirements. This provides a general framework for step-by-step system development (including iterative design and assessments studies), internal and external reviews and decision making.

At an early stage, only conceptual designs and assessments may be possible, reviews will be focussed on conceptual feasibility and acceptability, and the decisions may be about the direction of further studies. For example, at the current stage of the consideration of regional repositories for radioactive waste in the EU, the decision is whether or not to set up a European Development Organisation (EDO) to undertake further feasibility studies. If formed, that organisation would make studies that might lead to political decisions by Governments to participate or not in further steps.

At a later stage, as designs are developed, more specific feasibility and safety assessments will be possible, the reviews may become more technical, and the decisions may be made for the technical direction and regulatory control of the project. Ultimately, if the project continues to meet all the requirements of safety, security and socio-political acceptability, decisions about actual implementation will be reached.

Hence, Fig. 2.1 provides an overall vision for realising a waste management system based on safety and security, and implemented within socio-political and other constraints.

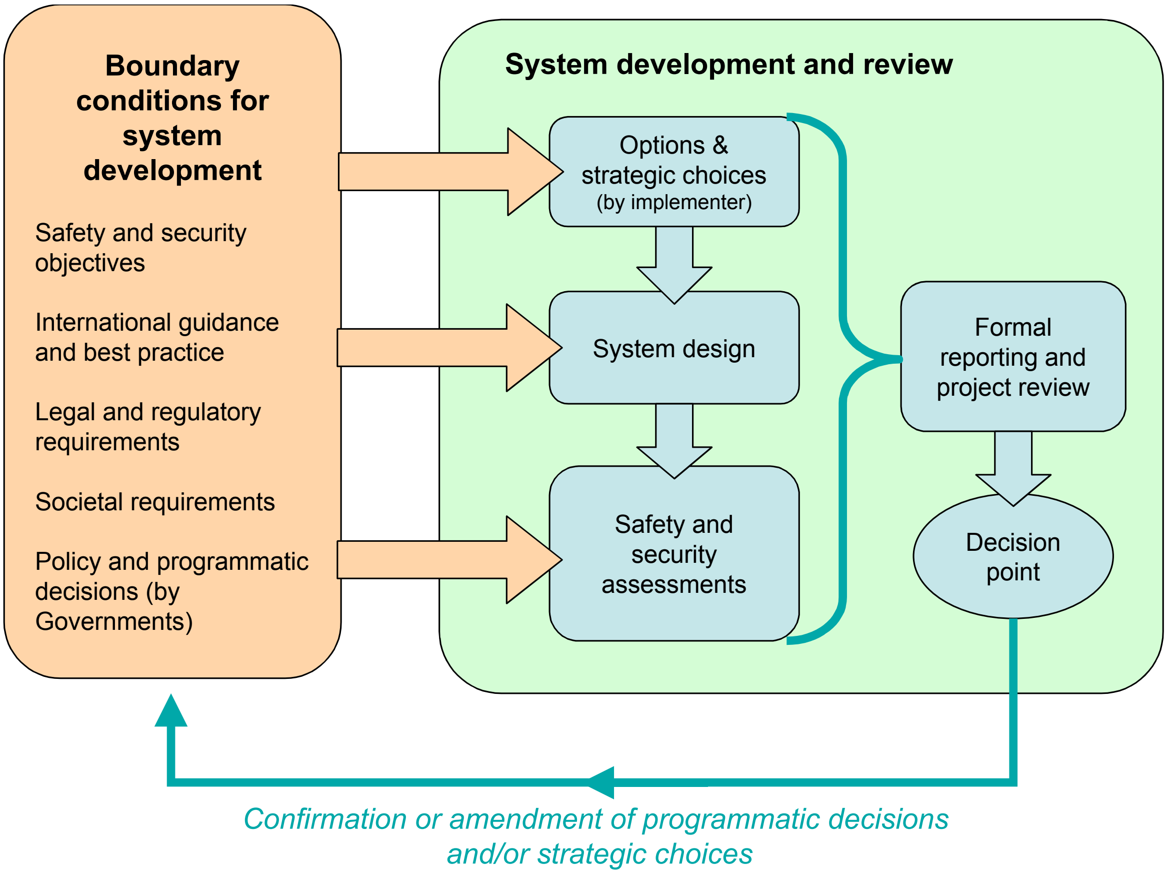

Fig. 2.1: Overview of system development and decision making focused on safety and security.

This report, under Work Package 4, focuses on the safety and security objectives and international guidance and best practice as they inform concept development and safety and security assessment. Other reports, under Work Packages 3 and 5 respectively, discuss legal requirements and societal requirements. Policy and programmatic decisions are beyond the scope of the current work, which is only to indicate the feasibility and possible paths forward from technical, legal and societal perspectives.

3 Boundary conditions for safety and security

This Chapter outlines the boundary conditions that must be met by any national or multinational system for the long-term management of radioactive waste in the EU. This includes obligations under international treaties and agreements and the internationally-developed objectives and principles related to safety, security and nuclear safeguards. Societal acceptance, political decisions and costs will also be important boundary conditions, but are not discussed in this report.

For the reasons explained in sections 2.1 and 2.2, the focus is on nuclear safety and security, and radiological protection.

3.1 International treaties and agreements on radioactive waste

There are a number of international treaties and agreements relevant to nuclear materials and radioactive waste. Those most relevant to a shared radioactive waste management system for high-level radioactive waste and spent fuel are as follows.

3.1.1 Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management

The Joint Convention on the Safety of Spent Fuel Management and the Safety of Radioactive Waste Management [10], administered by the International Atomic Energy Agency (IAEA), was opened for signature in September 1997. It has been ratified by, and is legally in force, in all EU countries.

The obligations of the parties are based on the fundamental principles of waste management, as described by the IAEA in the Safety Fundamentals publication, see section 3.2, which discusses safety objectives and principles.

Each and every contracting party is obliged to draw up and present a national report every three years, which is presented at the meeting of the fellow parties, e.g. [11]. These reports demonstrate the way in which the obligations under the Joint Convention are fulfilled. The reporting and discussion against the framework provided by the Joint Convention has the objective of encouraging a systematic improvement of the safety of the management of spent fuel and of radioactive waste at international level.

3.1.2 Treaty on the Non-Proliferation of Nuclear Weapons

The Treaty on the Non-Proliferation of Nuclear Weapons is an international treaty the objectives of which are to prevent the spread of nuclear weapons and weapons technology, to promote co-operation in the peaceful uses of nuclear energy and to further the goal of achieving nuclear disarmament. Opened for signature in 1968, the Treaty entered into force in 1970. A total of 187 parties have joined the Treaty, including the five “nuclear-weapon states” (i.e. the United States, Russia, the United Kingdom, France and China). The Treaty promotes co-operation in peaceful nuclear technology and access to this technology for all party states, while safeguards prevent the diversion of fissile material for weapons use.

The Treaty establishes a system of safeguards under the supervision of the IAEA that applies to nuclear facilities and nuclear material as identified under agreements between the IAEA and signatory states [12]. The IAEA is authorised to "establish and administer safeguards designed to ensure that special fissionable and other materials, services, equipment, facilities, and information … are not used in such a way as to further any military purpose" and to "apply safeguards, at the request of the parties, to any bilateral or multilateral arrangement, or at the request of a State, to any of that State's activities in the field of atomic energy". In practice this applies to all nuclear facilities in signatory states, except those that are specifically excluded, i.e. those with specific military purpose.

The EC operates a parallel system of nuclear safeguards inspection at civilian nuclear facilities in EU countries under the terms of the Euratom Treaty, see section 3.1.5.

Nuclear safeguards objectives and principles are discussed in section 3.4.

3.1.3 Convention on the Physical Protection of Nuclear Material

The Convention on the Physical Protection of Nuclear Material [13], also administered by the IAEA, was opened for signature in 1980 and has since been ratified in all EU countries. The aim of the convention is to avert the potential dangers posed by the unlawful taking and use of nuclear material. The Convention on the Physical Protection of Nuclear Material promotes international co-operation in the exchange of physical protection information and obligates parties to:

- make specific arrangements and meet defined standards of physical protection for international shipments of nuclear material;
- co-operate in the recovery and protection of stolen nuclear material;
- make as criminal offences specified acts to misuse or threats to misuse nuclear materials to harm the public;
- prosecute or extradite those accused of committing such acts.

As well as promoting consistent standards of nuclear security and protection, see section 3.3, the Convention places requirements related to information exchange and import, export and transit of nuclear materials to ensure that nuclear material will be protected during international transport.

In 2005, the IAEA issued an Additional Protocol in relation to terrorism [14], which strengthens the effectiveness and efficiency of the safeguards system as a contribution to nuclear non-proliferation objectives.

3.1.4 G8 Global Partnership Against the Spread of Weapons and Materials of Mass Destruction

The leaders of the G8 nations have made a number of statements and commitments related to nuclear non-proliferation [15]. In particular in 2002, in response to the terrorist attacks of September 11, 2001, and subsequent increased awareness of organised terrorism, the G8 Leaders launched the G8 Global Partnership Against the Spread of Weapons and Materials of Mass Destruction.

The G8 Leaders agreed on a set of six non-proliferation principles aimed at preventing terrorists or those who harbour them from acquiring or developing nuclear, chemical, radiological and biological weapons; missiles; and related materials, equipment or technology, and called on other countries to join in implementing these principles. In brief, the principles cover:

1. Promotion, implementation and strengthening of multilateral treaties and international instruments whose aim is to prevent the proliferation or illicit acquisition of such items.

2. Develop and maintain effective measures to account for and secure such items in production, use, storage and transport.

3. Develop and maintain appropriate effective physical protection measures applied to facilities which house such items.

4. Develop and maintain effective border controls, law enforcement efforts and international cooperation to detect, deter and interdict in cases of illicit trafficking in such items.

5. Development of effective national export and transhipment controls over items that may contribute to the development, production or use of nuclear, chemical and biological weapons and missiles.

6. Strengthening efforts to manage and dispose of stocks of fissile materials designated as no longer required for defence purposes, eliminate all chemical weapons, and minimize holdings of dangerous biological pathogens and toxins.

Under the initiative, the G8 nations will support specific cooperation projects, to address non-proliferation, disarmament, counter-terrorism and nuclear safety issues, and are committed to raising funds to support such projects. Initially, the priority is on the destruction of chemical weapons, the dismantlement of decommissioned nuclear submarines, the disposition of fissile materials and the employment of former weapons scientists. In future, the activities could be relevant to the development of shared facilities for safe and secure management of civil nuclear waste.

3.1.5 Treaties and directives of the European Union

All EU member states are signatories to the Euratom Treaty – one of the founding treaties of the European Union – and are bound to implement Directives of the European Union, including those under the Euratom Treaty, in national legislation.

The regulatory framework on the European level contains a number of important aspects related to radioactive waste management:

- The Euratom Safeguards provisions (Commission Regulation (Euratom) No 3227/76 of 19 October 1976 as amended [16]), that stipulate the provisions for the Member States to the security of fissile materials at the European level.
- The European directives concerning radiation protection (including especially Directive 96/29/Euratom [17]), that translate the ICRP Recommendations of Publication 60 and IAEA Basic Safety Standards (see section 3.2), into a European Directive.
- The European Directive 98/83/EG on the quality of water for human consumption, imposing a dose limit of 0.1 mSv per year on drinking water for all radionuclides except tritium, potassium-40, radon and its decay products.

The Resolution of the European Council of 19 December 1994, concerning the management of radioactive waste, states a number of principles for radioactive waste management, although these are not legally binding on the member states. These principles include that each member state is responsible for ensuring that the radioactive waste on its territory is properly managed, and for establishing suitable facilities for the treatment, conditioning, storage and disposal of radioactive waste. Continued co-operation with various international bodies, to provide international guidance and standards for the safe management of radioactive waste and to encourage the adoption of best available techniques and best environmental practice, is encouraged.

3.2 Safety objectives and principles

The International Commission on Radiological Protection (ICRP) has developed an evidence-based and formal system of radiological protection, as set out in Publication 60 [18] now superseded by Publication 103 [19]. The system set out in Publication 60 has been adopted into international principles and requirements for the safe use of nuclear energy and of radioactive materials as promulgated by the International Atomic Energy Agency (IAEA). This includes Basic Safety Standards for protection against ionizing radiation and the safety of radiation sources [20] (which, in turn, have been incorporated into Directives under the Euratom Treaty [17] and also national regulations).

Both the ICRP system of radiological protection and the IAEA system of safety standards have changed in detail over the years, and have also grown in extent as different situations and radiological activities have been considered. The core principles have, however, remained remarkably unchanged. The recent IAEA Fundamental Safety Principles [21], sets down the safety objective and high-level safety principles that apply to all activities related to peaceful uses of nuclear energy and radioactive materials. The safety objective and principles are reproduced in Box 3.1.

The ICRP system of radiological protection was developed primarily to apply to the exposure of workers (occupational exposure), members of the public and medical exposures in which the sources of radiation are under active control. The system has been extended by the ICRP to cover circumstances of potential exposures, including exposures that may occur in the future as a result of disposal of solid radioactive waste [22, 23].

The ICRP recommendations provide basic protection objectives and criteria that are incorporated into IAEA safety requirements. These also set down the requirements that must be met to ensure safety for various practices involving nuclear processes and radioactive materials and wastes. Most relevant to the SAPIERR-2 project are the safety requirements related to:

- transport of radioactive materials [24];
- predisposal management (which includes waste processing and storage) [25];
- geological disposal of radioactive waste [26].

The IAEA safety requirements contain conditions that the IAEA consider “shall” be met in order to assure nuclear and radiological safety. Beneath these are a series of safety guides that discuss conditions that “should” be met to accord with best practice. For example, there are several safety guides related to aspects of storage facilities for radioactive waste [27, 28, 29]. IAEA technical documents, or TECDOCs, discuss specific issues of interest to member states.

Box 3.1: IAEA safety objective and safety principles

Reproduced from the IAEA Fundamental Safety Principles [21].

SAFETY OBJECTIVE

The fundamental safety objective is to protect people and the environment from harmful effects of ionizing radiation.

This fundamental safety objective of protecting people – individually and collectively – and the environment has to be achieved without unduly limiting the operation of facilities or the conduct of activities that give rise to radiation risks. To ensure that facilities are operated and activities conducted so as to achieve the highest standards of safety that can reasonably be achieved, measures have to be taken:

(a) To control the radiation exposure of people and the release of radioactive material to the environment;

(b) To restrict the likelihood of events that might lead to a loss of control over a nuclear reactor core, nuclear chain reaction, radioactive source or any other source of radiation;

(c) To mitigate the consequences of such events if they were to occur.

SAFETY PRINCIPLES

Principle 1: Responsibility for safety – The prime responsibility for safety must rest with the person or organization responsible for facilities and activities that give rise to radiation risks.

Principle 2: Role of government – An effective legal and governmental framework for safety, including an independent regulatory body, must be established and sustained.

Principle 3: Leadership and management for safety – Effective leadership and management for safety must be established and sustained in organizations concerned with, and facilities and activities that give rise to, radiation risks.

Principle 4: Justification of facilities and activities – Facilities and activities that give rise to radiation risks must yield an overall benefit.

Principle 5: Optimization of protection – Protection must be optimized to provide the highest level of safety that can reasonably be achieved.

Principle 6: Limitation of risks to individuals – Measures for controlling radiation risks must ensure that no individual bears an unacceptable risk of harm.

Principle 7: Protection of present and future generations – People and the environment, present and future, must be protected against radiation risks.

Principle 8: Prevention of accidents – All practical efforts must be made to prevent and mitigate nuclear or radiation accidents.

Principle 9: Emergency preparedness and response – Arrangements must be made for emergency preparedness and response in case of nuclear or radiation incidents.

Principle 10: Protective actions to reduce existing or unregulated radiation risks – Protective actions to reduce existing or unregulated radiation risks must be justified and optimized.

3.3 Security objectives and principles

The Convention on the Physical Protection of Nuclear Material [13], see section 3.1.3, promotes international co-operation and common security standards for nuclear facilities and materials, and is intended primarily to protect nuclear facilities and nuclear material from unauthorised and malicious acts.

The principles and implementation requirements are set out in the IAEA document on the Physical Protection of Nuclear Material and Nuclear Facilities [30]. The principles and requirements are designed to minimise the opportunities for theft of nuclear materials or sabotage of nuclear facilities. They apply to the use, storage and transport of materials containing fissile isotopes of uranium and plutonium, including spent nuclear fuel.

Nuclear material is categorised according to its fissile potential from category I, the highest risk material, to category III, the lowest risk material considered.

- Category I is fissile material in sufficient quantity and form to be useful in producing a workable nuclear device, for example more than 2 kg of unirradiated plutonium.

- Category II material that either in total mass or need for further operation is not useful in itself for producing a nuclear device, for example spent nuclear fuel.

- Category III material in quantity or quality that is insufficient in itself for producing a nuclear weapon, for example small amounts (less than 500g) of unirradiated plutonium.

Amounts of unirradiated plutonium or uranium less than 15g, and natural uranium, fall outside the categorisation, but should be protected at least in accordance with prudent management practice.

Principles of physical protection are realised through administrative and technical measures, including physical barriers. Each country should evaluate the level of threat it faces and produce a document that defines the ‘Design Basis Threat’ (DBT), see section 6.2.1. The security measures necessary to protect against the DBT are then devised for specific nuclear facilities. Protection can be achieved through:

- hardware, e.g. alarms or physical barriers such as security gates and fences;
- personnel, e.g. the use of guards;
- procedures, e.g. security vetting, controlling access to the facility, security of computer systems;
- facility design and layout, e.g. layout to ensure possibilities for isolation of sensitive areas and physical barriers.

The IAEA Physical Protection of Nuclear Material and Nuclear Facilities [30] sets Requirements for physical protection:

- against unauthorised removal of nuclear material in use and storage;
- against sabotage of nuclear facilities and nuclear material during use and storage;
- and of nuclear material during transport

These are discussed further in section 6.1.

3.4 Nuclear safeguards objectives and principles

Safeguards, established under the Treaty on the Non-Proliferation of Nuclear Weapons, see section 3.1.2, are to prevent the diversion of fissile material for weapons use primarily by states or their agents.

The principles and implementation requirements are set out in the IAEA Safeguards System [31]. The requirements apply to fissile nuclear materials supplied by the IAEA under project agreements or as submitted to safeguards by signatory states of the Treaty.

The IAEA may inspect safeguarded nuclear materials and principal nuclear facilities to verify compliance with safeguards agreements and to assist States in complying with such agreements and in resolving any questions arising out of the implementation of safeguards. Inspections may include, as appropriate:

- audit of records and reports;
- verification of the amount of safeguarded nuclear material by physical inspection, measurement and sampling;
- examination of principal nuclear facilities, including a check of their measuring instruments and operating characteristics; and
- check of the operations carried out at principal nuclear facilities and at research and development facilities containing safeguarded nuclear material.

The IAEA will review the design of principal nuclear facilities, for the purpose of satisfying itself that a facility will permit the effective application of safeguards. To facilitate this, the State must submit relevant design information, including information on characteristics of the facility as may bear on safeguards procedures.

The state shall arrange for the keeping of records with respect to principal nuclear facilities and all safeguarded nuclear material outside such facilities, and this system of records must be agreed with the IAEA.

The State shall report to the IAEA without delay: if any unusual incident occurs involving actual or potential loss, destruction, or damage to, any safeguarded nuclear material or principal nuclear facility; or if there is good reason to believe that safeguarded nuclear material is lost or unaccounted for in quantities that exceed the normal operating and handling losses that have been accepted as characteristic of the facility.

This report does not address non-proliferation or nuclear safeguards in detail, since the controls are equally applicable to shared or national nuclear activities and under the same internationally-supervised arrangements. A shared repository programme does, however, offer potential advantages from non-proliferation and nuclear safeguards perspectives and these are mentioned in section 6.4.2.

3.5 Implementation – best practice and regulatory supervision

A regional waste management system would have to meet standards of practice for design, assessment and implementation at least as good as established in other EU countries.

It can be expected that, at the time of beginning implementation of regional facilities, several high-quality long-term interim stores and some geological disposal facilities might be already operating or under construction in the EU. The design, assessment and implementation of regional facilities will take advantage of this experience.

A shared radioactive waste management system would be subject to all the requirements of the treaties and agreements, as discussed in sections 3.1, and meet standards as discussed in sections 3.2 to 3.4, through the obligations of the states in which radioactive waste originates, through which waste is transported, and in which the shared facilities are sited.

Shared facilities and transport shipments would have to comply with the relevant regulations, obtain necessary licenses and be subject to regulatory supervision according to the laws of the hosting country. It is possible that the EU may offer specific guidance and/or oversight for a shared project.

National regulations are put in place consistent with obligations under international treaties and agreements, and EU laws, and taking account of specific national requirements and practices. Any country that volunteered to host a regional facility would have to possess a suitably developed national regulatory system, and would need to develop regulations and guidance specific to the proposed facility or facilities. The EC, IAEA and project partner countries would most likely be directly involved in this process.

Requirements on transport are already well harmonised, since IAEA transport regulations are designed to be incorporated in their entirety into national regulations, and have been so incorporated in all EU countries.

4 Regional disposal system – options and scenarios

This chapter sets out a brief description of the options and scenarios for a shared regional disposal system consisting of one or more deep geological repositories, encapsulation facilities, and transport and transfer arrangements. This includes consideration of inventory, number of repositories, host rock options, transport distances, timing and operational steps.

The options and scenarios are based on those developed in SAPIERR I [4] and consistent with those considered in the SAPIERR II report on economic aspects [32].

4.1 Regional or shared disposal system concept

The radioactive waste management system considered is the complete chain of activities and facilities that would be needed take radioactive waste safely and securely from storage facilities at nuclear power plants, or from centralised national storage facilities, to final disposal in one or more regional deep geological repositories.

The radioactive wastes considered are the same as in SAPIERR I, that is spent nuclear fuel (SF) from commercial power plants, high-level waste (HLW) from the reprocessing of spent fuel and long-lived low and intermediate-level radioactive waste (LILW-LL), not suited for disposal in near-surface facilities.

A shared European disposal system would thus consist of:

- one or more deep geological repositories for final disposal;
- one or more encapsulation facilities for encapsulation of HLW/SF (although encapsulation might also be offered on a commercial basis by a country or countries not involved in the shared repository);
- buffer storage facilities at each repository and encapsulation plant (although a centralised shared storage facility could also be an option);
- transport and transfer arrangements between national storage facilities, the shared encapsulation facilities and shared geological repositories.

For LILW-LL, it is assumed that the waste would be conditioned (processed and packaged) at national facilities into waste packages suited to storage and disposal. LILW-LL packages may be transported in disposable overpacks or in re-usable transport containers if additional shielding or containment is required for transport.

A range of possibilities can be considered for the number and location of shared facilities. For example, Figure 4.1 illustrates options within the general concept of a shared management system for HLW/SF leading to disposal in a single geological repository. The figure shows options with and without a regional storage facility and options for co-siting or independent siting of the facilities.

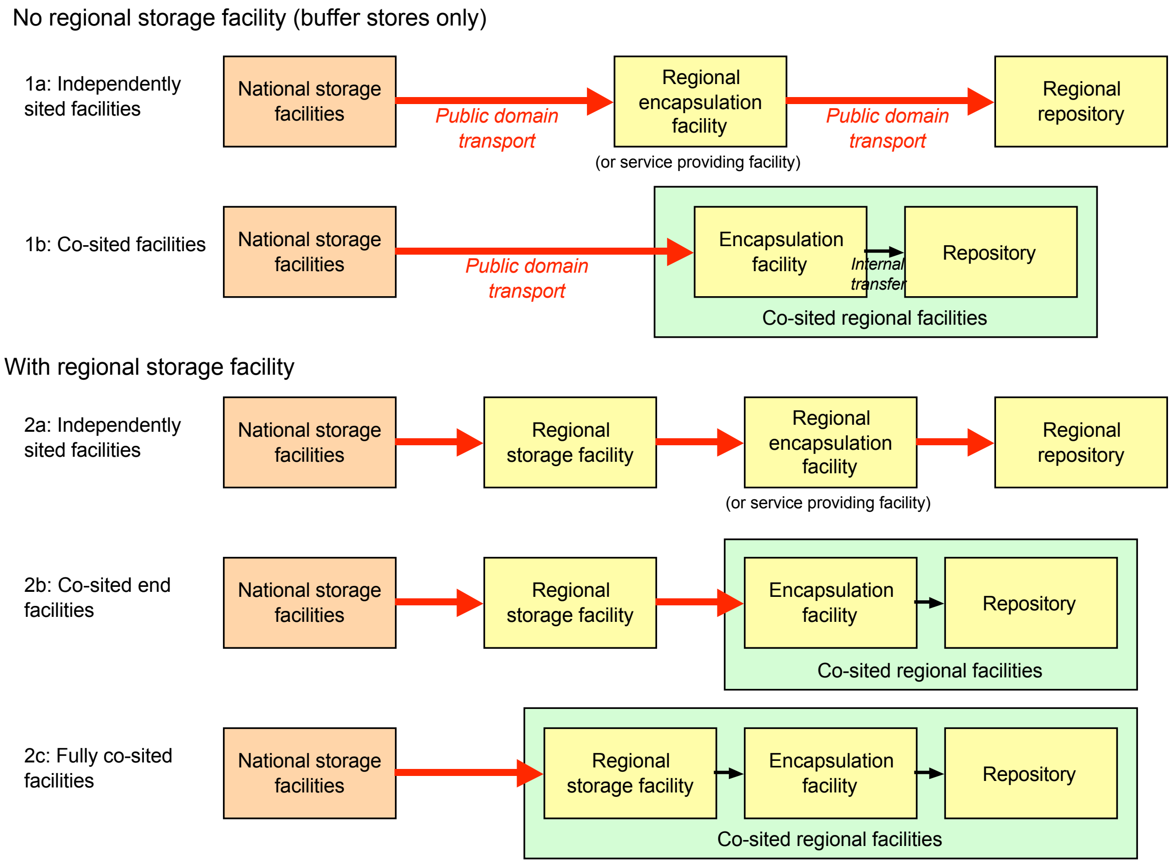

Figure 4.1: Example options for a shared radioactive waste management system

SAPIERR I considered the issue of shared storage facilities and came to the conclusion that, provided disposal facilities were available in a timely fashion consistent with the rate of waste arising and becoming ready for disposal, then there was little advantage in centralised shared stores. Adequate storage is available at the source of the wastes and, with modular dry cask storage, there is no large economy of scale impact. However, buffer storage capacity would be needed at the locations of the encapsulation plant and repository.

In SAPIERR II, and for the purpose of considering safety and security aspects, we focus on Option 1 indicated in Figure 4.1, i.e., without a dedicated regional storage facility.

A set of scenarios or options for implementation of regional repositories in the EU has been defined for the purposes of the analysis as described in the following sections. These can then be compared against a base case, in which each country is constrained to have a national geological disposal facility. The scenarios consider possible variations with respect to:

- the inventory of waste for disposal in a shared repository or repositories;
- potential repository host rocks, and the number and arrangement of repositories;
- transport routes and distances arising out of different arrangements of repositories.

4.2 Inventory of waste

Although a particular waste inventory is not central to the viability of a shared European repository, some assumptions are needed as a basis for exploring the sharing concept. In the SAPIERR II study we have considered a 'large' European inventory and a 'small' European inventory.

4.2.1 The 'large' inventory situation

In SAPIERR I, the waste inventory used as a reference was the total waste arisings from the fourteen countries from which organisations participated in the project [5]. This was not meant to indicate that any of these countries had chosen a final disposal strategy, but rather to give quantitative working assumptions. For ease of comparison with SAPIERR I, the same inventory is used as the reference case in the current study, again emphasising that the inclusion of a national inventory within the SAPIERR reference inventory does not imply that the country concerned would choose to participate in a shared European solution.

The reference inventory in SAPIERR I at 2040 was derived as:

- 25,637 t of spent fuel (SF);
- 355 m^3 of vitrified high level wastes (HLW);
- 31,000 m^3 of long-lived intermediate level wastes (LILW-LL).

SAPIERR I considered that the HLW/SF could be packaged for disposal in about 13,500 containers (approximately 13,200 for SF and 300 for HLW).

For the 'large' inventory, SAPIERR II uses the waste tonnages and volumes above but we again emphasise their arbitrary nature – an eventual European regional repository could hold more or less waste than considered here.

4.2.2 The 'small' inventory situation

To look at the economics and also the safety aspects of a situation where only two or three countries decide to share disposal solutions, we have also looked at a 'small' inventory.

The 'small' inventory is derived from an evaluation of the individual national inventories of the fourteen SAPIERR I countries and comprises approximately 25% of the 'large' inventory:

- 6280 t of spent fuel (this equates to about 3500 containers for disposal);
- 6800 m^3 of long-lived intermediate level wastes (LILW-LL).

To arrive at these figures, we looked at a range of hypothetical, 2 and 3-country sharing situations that gave total amounts of spent fuel of between about 4700-7600 t SF and between 6200-9000 m^3 of LILW-LL, with the numbers actually selected for the 'small' inventory model being averages of the various situations considered.

Inventories in these ranges could be derived if, for example: Belgium and the Netherlands were to share a disposal solution; Bulgaria and Romania were to share a solution; Slovakia, Slovenia and the Czech Republic were to share a solution. These hypothetical partnerships do not reflect in any way on the intentions or policies of these countries but are provided as illustrations of scale for the 'small' inventory situation.

4.3 Repository host rocks and number of repositories

Since we are not considering location in a particular country or countries, there is no reason to constrain the geological environment in which a regional repository might be located. It is most likely, however, to be located in a host rock that has been investigated with respect to potential for hosting a geological repository in Europe. The investigated options, e.g. see [33], include:

- strong fractured rocks (e.g. granite, gneiss, basalt etc.);
- argillaceous sediments of varying degrees of induration (e.g. clay, mudstone, shale etc.);
- evaporites (e.g. halite (rock salt), anhydrite etc.).

In SAPIERR I, repositories were considered to be in either 'hard rocks' or 'sediments' mainly on the basis that model cost information was most readily available for these concepts. In SAPIERR II, six scenarios are considered with respect to economic aspects [34], as listed in Table 4.1.

These consider co-siting of disposal for HLW/SF and LILW-LL, separate siting of HLW/SF and ILW repositories and two separate repositories each accommodating one half of the total SAPIERR waste inventory. There are various reasons why a 'two repository' scenario might be favoured. These include efficiency with respect to transport, ensuring security of supply of disposal services and catering for differing times of waste arising in different countries.

Economic scenario	Description
Scenario I(H)	Repository for all wastes at a single site in hard rock.
Scenario I(S)	Repository for all wastes at a single site in sediments (clays).
Scenario II(H)	Separate repositories for HLW/SF and for LILW-LL, each in hard rock.
Scenario II(S)	Separate repositories for HLW/SF and for LILW-LL, each in sediments (clays).
Scenario IIIa:	Two separate repositories, each with half the SAPIERR waste inventory, one in hard rock and one in sediment (each has its own encapsulation plant).
Scenario IIIb:	The same as Scenario IIIa, but with only one encapsulation plant, located at the hard rock repository site.

Table 4.1: Scenarios for regional repositories considered in economic analyses [34]

In principle, any repository for long-lived radioactive waste in the EU will have to meet common standards of safety and security stemming from international guidance and EU laws. In addition, in the absence of information on potential sites it is not possible to resolve most differences in safety or security that might arise from the differences implied in the scenarios indicated in Table 4.1. Hence, for the purpose of discussion of safety and security we consider a smaller set of cases as indicated in Table 4.2.

Safety and security case	Inventory, repository and encapsulation	Host rock
Case I	Large inventory: Single repository for SF/HLW and LILW-LL and co-sited encapsulation plant for SF/HLW.	In hard rock, argillaceous sediments or evaporite
Case II	Large inventory: Separate repositories for SF/HLW, with co-sited encapsulation plant, and for LILW-LL.	In hard rock, argillaceous sediments or evaporite
Case IV	Small inventory: Single repository for SF/HLW and LILW-LL and co-sited encapsulation plant for SF/HLW.	In hard rock, argillaceous sediments or evaporite

Table 4.2: Cases for regional repositories considered with respect to safety and security

Safety and security cases I and II correspond to economic scenarios I and II; case IV considers the "small inventory", see section 4.2.2.

It is not the intention to separately assess the safety and security of these options, but rather to point to possible differences or trends between the options with respect to safety and security, and also relative to a national scenario in which each country must develop a national geological disposal facility. In the event in this study, most attention with regard to safety has been focused on spent nuclear fuel, so that quantitative estimates of radiological impact, see section 5.6, are only made for Cases I and IV. Security aspects are discussed in qualitative terms, wherein common issues arise, such that the cases are not strongly differentiated.

4.4 Transport routes and distances

Transport routes are not considered to be a decisive legal or technical factor, since radioactive wastes in the EU, like any other goods, must be able to be freely shipped across Europe. In practice, it might be a political or societal problem if wastes had to cross countries that were not involved in the shared disposal solution and were anti-nuclear.

The distances and routes over which waste has to be transported in national or multinational scenarios, and the transport modes, are important when comparing safety, security and environmental aspects. Transport routes and distances cannot be firmly estimated prior to identifying the participating countries and possible locations of shared facilities. In the absence of this information, the following arbitrary, but plausible assumptions are made.

For security reasons, and practicality of moving heavy loads, rail transport might be preferred wherever the existing infrastructure allowed. Dedicated railhead facilities could be constructed at encapsulation and disposal facilities. Road transport could be used where needed to fill 'gaps' between waste sources and national rail networks.

As approximate estimates for the average distance to be transported, we can look at the area of countries and use the radius of the equivalent circles. The average area of countries in Europe (total area of the EU divided by the number of countries) is ~156,400 km^2, giving an average country radius of 220 km. To take into account the fact that countries interested in a multinational facility are more likely to be the geographically smaller countries, and considering that in a national scenario the incentive will be to site a repository near to existing nuclear facilities if possible, this number is rounded down to a reference value of 100 km. If the whole area of Europe were joined, the radius would be 1240 km. If divided into two equal area regions (i.e. the two repository scenario) the radius for each is 880 km. For the small inventory case, we can assume 2 to 3 adjacent or nearly adjacent countries within a region of area 450,000 km^2 and hence transport radius of 380 km.

Rounding these numbers, we adopt the following indicative transport distances for waste shipments.

National repositories only scenario	100 km
Multinational scenarios I and II	1200 km
Multinational scenario III	900 km
Small inventory case IV	400 km

These figures assume that encapsulation is in all cases at the repository site(s). If disposal were national but encapsulation offered internationally as a service, the transport distances in the national scenario would increase significantly.

4.5 Timing of encapsulation and disposal

SAPIERR I sketched out a timetable leading to disposal operations between 2035 and 2095, that is an operating period of 60 years. This assumption impacts upon the storage requirements until these dates (as discussed in section 4.1) and thus on issues such as security at existing and future stores. It also constrains the time available for siting work and associated R&D, which both have significant impacts on estimated costs.

One of the key features of the shared repository concept is that the volumes of wastes for disposal held in national stores will more quickly accumulate to a total at which it is economic to begin operation of geological repository. By contrast (within the national repositories only scenario), a key factor in relation to safety and security is the implied longer time that waste may have to remain in national stores pending the construction and operation of national repositories.

If a clear commitment to shared repositories is in place, but construction and operation of such repositories is delayed, then there may be a case for considering shared regional storage facilities to improve safety and security in the interim.

Our consideration of repository operations, see section 5.6.4 and Table 5.7, lead us to assume a repository operating period of 50 years for the large inventory and 25 years for the small inventory, but this has only a minor impact on the calculated results.

4.6 Operational steps

Finally, in order to discuss safety and security it is helpful to set down the operational steps that will be needed to take waste from national stores to its final emplacement in a geological repository.

As discussed in the previous sections, there are various options with respect to inventory, potential repository host rocks, number and location of facilities (storage, encapsulation and disposal facilities), transport arrangements and timing. In principle, however, the operational steps are relatively common with the main possible differences related to the relative locations of shared facilities (co-sited or independently sited) and timing and logistics, which may imply requirements for storage between steps. The main steps are:

- withdrawal of waste from national storage facilities;
- transport in the public domain from national storage facilities to the regional encapsulation facility for SF/HLW, or directly to the repository for conditioned (treated and packaged) LILW-LL;
- receipt of SF/HLW at shared encapsulation facility and encapsulation in disposal containers;
- transport of SF/HLW in the public domain from encapsulation to disposal facility (this step is omitted if the facilities are co-sited);
- receipt of SF/HLW in disposal containers and conditioned LILW-LL at shared repository, and emplacement in the underground.

Details of the process, e.g. types of transport container and buffer store arrangements, could be flexibly arranged to match the source and condition of waste and logistics of the encapsulation and disposal facilities. As an illustration, Figure 4.2 sets out a possible process for passage of SF though the 5 steps indicated above. SF is selected as the dominant waste quantitatively (in terms of the number of shipments needed) and the most sensitive from safety and security perspectives. Additional and alternative options and possibilities with respect to SF, and with respect to HLW and LILW-LL, are discussed in the following chapters.

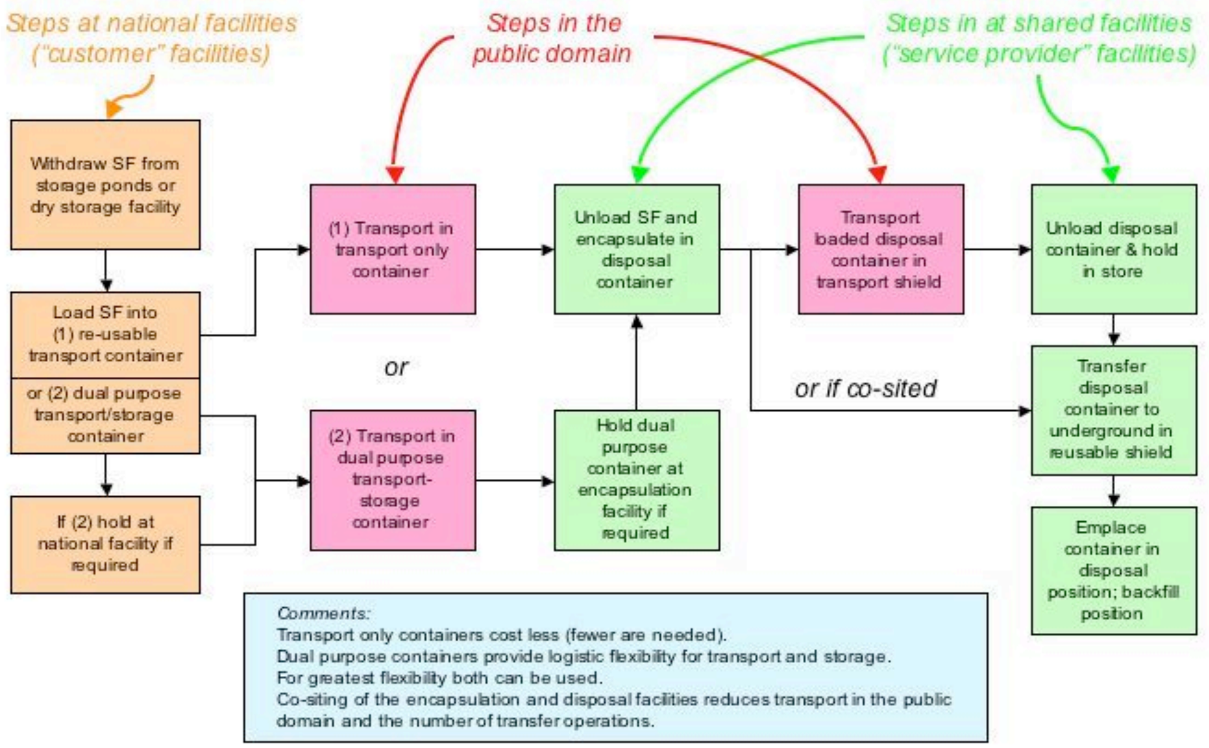

Fig. 4.2: Steps in the movement of spent fuel from national stores to emplacement in a shared repository

5 Safety

This chapter discusses nuclear and radiological safety aspects of radioactive waste management from waste acceptance to disposal. This includes discussion of safety standards, waste and waste package acceptance, transport safety, operational safety of facilities, and repository post-closure safety. Proportionately, most attention is directed at safety related to spent nuclear fuel, which is, quantitatively and radiologically, the dominant waste form.

Sections 5.1 to 5.5 describe how safety is achieved throughout all stages in a generic disposal programme. This leads to the general conclusion that a shared European repository can also be sited, constructed and operated in a manner that ensures adequate safety. Section 5.6 then provides a safety overview and addresses the important question of whether the overall radiological impact in Europe will be altered if a large number of national repositories are replaced by fewer shared facilities. The comparison is done by presenting indicative estimates of radiological impacts for two notional shared waste management systems and national systems of equivalent capacity. Section 5.7 makes some final remarks and conclusions on safety.

5.1 Safety standards

5.1.1 International safety standards

Radioactive waste management systems, including geological disposal facilities, are required to be developed in such a way that human health and the environment are protected both now and in the future [21]. In IAEA guidance, and here, the prime concern is the radiological hazard presented by radioactive waste.

Operations at facilities and transport

The international radiological protection requirements and criteria for radioactive waste transport, storage and encapsulation operations, and for the operational period of a geological disposal facility, are the same as for any licensed nuclear facility, and are established in the IAEA Basic Safety Standards [20]. During these steps, the source is under control, releases can be verified, exposures can be controlled and actions can be taken if necessary. The primary goal is to ensure that radiation doses are as low as reasonably achievable. A necessary, though not in itself sufficient, condition is that all doses are kept within applicable dose limits, see Box 5.1.

The optimisation of protection (that is, ensuring that radiation doses are as low as reasonably achievable) is required to be considered in the design of the waste management facilities and in the planning of operations above and below the ground.

An operational radiation protection programme is required to be in place to ensure that the doses to workers during normal operations are controlled and that the requirements for the limitation of radiation doses are met. In addition, contingency plans are required to be in place for dealing with accidents and incidents and for ensuring that any consequent radiation doses are controlled to the extent possible with due regard to the relevant emergency reference levels [35].

Box 5.1: Dose limits for occupational exposure and for members of the public

From IAEA Basic Safety Standards [20], Schedule II.

The occupational exposure of any worker shall be controlled so that the following limits be not exceeded:

(a) an effective dose of 20 mSv per year averaged over five consecutive years;

(b) an effective dose of 50 mSv in any single year. [2]

The estimated average doses to the relevant critical groups of members of the public that are attributable to practices shall not exceed the following limits:

(a) an effective dose of 1 mSv in a year;

(b) in special circumstances, an effective dose of 5 mSv in a single year provided that the average dose over 5 consecutive years does not exceed 1 mSv per year. [2]

Transport-specific requirements

The doses and risks associated with the transport of radioactive waste to an encapsulation plant and/or geological disposal facility would be managed in the same way as for the transport of other radioactive material. That is all radioactive transport operations must comply with the IAEA Regulations for the Safe Transport of Radioactive Material [24]. In particular, a radiation protection programme shall be established for the transport of radioactive material. For occupational exposures, where it is assessed that the effective dose:

- is likely to be between 1 and 6 mSv in a year, a dose assessment programme via workplace monitoring or individual monitoring shall be conducted;
- is likely to exceed 6 mSv in a year, individual monitoring shall be conducted.

Dose limits for occupational exposure and for members of the public are as given in Box 5.1.

A fundamental feature of the IAEA transport regulations is that safety of transport of larger amounts of radioactive material is vested in the transport package design. All shipments of SF/HLW and LILW-LL (hereafter abbreviated as ILW) will take place in IAEA “Type B” waste packages [24]. Such packages are designed to provide adequate shielding and complete containment of the waste in normal transport conditions, and to remain intact with minimal potential for radioactive release in accident conditions, see section 5.3.

Repository post-closure

To ensure the long-term (post-closure) radiological safety of a geological repository presents a special challenge because the source is no longer under direct control and it cannot be expected that a radiation protection programme will be in place at the time when radionuclide releases from the repository may occur.

The IAEA safety requirements document for geological disposal [8] sets out the objective and criteria for this case as follows:

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2 In addition, specific criteria apply to certain tissues – the lens of the eyes, extremities and skin.

Objective – Geological disposal facilities are to be sited, designed, constructed, operated and closed so that protection in the post-closure period is optimised, social and economic factors being taken into account, and a reasonable assurance is provided that doses or risks to members of the public in the long term will not exceed the dose or risk level that was used as a design constraint.

Criteria - The dose limit for members of the public from all practices is an effective dose of 1 mSv in a year, and this or its risk equivalent is considered a criterion not to be exceeded in the future. To comply with this dose limit, a geological disposal facility (considered as a single source) is designed so that the estimated average dose or average risk to members of the public who may be exposed in the future as a result of activities involving the disposal facility does not exceed a dose constraint of not more than 0.3 mSv in a year or a risk constraint of the order of 10^{-5} per year [3].

5.1.2 National regulatory standards

The IAEA Basic Safety Standards are included in the EC Directive 96/29/Euratom and hence implemented in national laws in the EU, see section 3.1.5. National governments and/or regulatory bodies consider these as basic standards but may make more stringent or specific requirements consistent with their national situation and preferences.

For example in the UK, the regulator has set a dose constraint of 0.3 mSv per year to a representative member of the critical group for a geological disposal facility during the period of authorisation[4], and 0.5 mSv per year from the site at a whole [36]. That is a fraction of the 1 mSv per year of the IAEA Basic safety Standards, to account that other sources may also contribute, plus a condition for the case of a repository developed from a nuclear site that would itself constitute a source. For the period after authorisation, the assessed risk from a disposal facility to a person representative of those at greatest risk should be consistent with a risk guidance level of 10^{-6} per year. That is a fraction of the risk constraint of the order of 10^{-5} per year set in the IAEA safety requirements, to account for uncertainty at long times in the future, but also noting that the guidance level indicates a level of environmental safety that the regulator is looking for, not an absolute requirement that must be met.

Ferch [37] and a recent Nuclear Energy Agency report [38] have surveyed national dose and risk criteria for the long-term performance geological repositories. Table 5.1 illustrates the variation between some different European countries in the regulatory target values that would be applied to a national repository. It should be observed, however, that crucial factors are the interpretation of the constraint or reference level by the regulator and conditions on how the dose or risk values should be calculated, which may also vary between countries.

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3 It is recognised that radiation doses to individuals in the future can only be estimated and that the uncertainties associated with these estimates will increase for times further into the future. Care needs to be exercised in using the criteria beyond the time where the uncertainties become so large that the criteria may no longer serve as a reasonable basis for decision making.

4 The period of authorisation would include the period of operations, plus the period after disposals are completed up to completion and closure of the repository, and including any further period of control that the regulator considers necessary.

Country	Dose constraint or reference level	Risk constraint or reference level
Belgium	0.1 to 0.3 mSv y^{-1} (Working value)	10^{-5} y^{-1} (Working value)
Czech Republic	0.25 mSv y^{-1}	Scenarios with $P < 10^{-6}$ need not be considered
Finland	0.1 mSv y^{-1} for normal evolution	Unlikely events assessed against the risk equivalent of 0.1 mSv y^{-1}
France	0.25 mSv y^{-1} for normal evolution	Case by case judgement
Germany	0.3 mSv y^{-1} (Criteria under revision)	-
Hungary	0.1 mSv y^{-1} for normal evolution	10^{-6} y^{-1} for disruptive events
Netherlands	0.1 mSv y^{-1} dose limit 0.04 mSv y^{-1} optimisation goal	
Slovakia	0.1 mSv y^{-1} for normal evolution	-
Spain	0.1 mSv y^{-1} for high P scenarios	10^{-6} y^{-1} for low P scenarios
Sweden	-	10^{-6} y^{-1}
Switzerland	0.1 mSv y^{-1} for likely scenarios	10^{-6} y^{-1} for unlikely scenarios
UK	-	10^{-6} y^{-1} reference level

Table 5.1: Regulatory target values that would be applied to a national repository in different European countries, from [37]

5.1.3 Project standards

Having considered the actual operations and expected radiological impacts based on precedent of similar operations conducted using best practice, the developer of the waste management system may set even more stringent reference levels as design targets or goals for the system.

For example UK Nirex Ltd., which was formerly responsible for development of deep geological repository concepts in the UK, set both design limits (corresponding to dose constraints), and design targets for doses to worker and members of the public for use in assessment of operational safety [39], see Table 5.2.

For normal operations		Design limit	Design target
Occupational dose assessment	Exposed workers	20 mSv y^{-1}	2 mSv y^{-1}
	Other workers	5 mSv y^{-1}	0.5 mSv y^{-1}
Off-site dose assessment	Members of the public	1 mSv y^{-1}	0.2 mSv y^{-1}

Table 5.2: Example of project standards for operational safety related to geological disposal, from [39]

The setting of design targets that are more stringent than legal or regulatory requirements can be seen as part of the optimisation process.

Thus, the international safety standards as described in section 5.1.1 can be viewed as the minimum safety goals for the development of a shared waste disposal system. More stringent requirements may be placed by national laws or by the regulatory bodies, and the EDO may set design targets for radiological protection consistent with what is achievable with best practice for the various facilities and activities.

5.2 Waste acceptance and package acceptance

A key safety feature of any management system for radioactive waste are the conditions for acceptance of waste into the system, termed Conditions For Acceptance (CFA) or Waste Acceptance Criteria (WAC), and the controls and procedures to ensure that the required conditions and criteria are met.

A key task for the EDO will be to set out CFA or WAC consistent with the characteristics of the proposed shared waste management facilities so as to ensure that all necessary transport, handling, storage and disposal steps can be safely accomplished and that the long-term safety of disposal can be assured. The EDO will also set out at least the general plan by which waste providers will demonstrate compliance with the CFA/WAC. The ERO will further detail the procedures for providing information on the waste and assuring that the characteristics fully meet the CFA/WAC. The ERO may also consider the case of waste packages or types that do not fully meet the standard CFA/WAC, and assess their acceptability or need for reworking, repackaging or over-packing, on a case by case basis so as not to unreasonably refuse any proffered waste type from the project members.

The primary requirement is for accurate and verifiable information concerning the waste, e.g. fuel type, irradiation history, and post-irradiation handing and storage, inventory and condition. Handling facilities within the shared facilities will be designed with flexibility to cope with a range of fuel and other waste types and packages, but this flexibility cannot be open ended. Standardised waste packages are particularly important for ILW; although a range of ILW waste types and forms may be acceptable from storage and disposal perspectives, standardising on a limited number of outer envelope designs facilitates efficient handling during transport and during emplacement in the repository.

The EDO/ERO will also specify requirements on transport packages, road and rail vehicles, and arrangements for shipments to ensure all conditions needed to assure safety during transport in the public domain are in place.

The extent to which, and/or point at which, the ERO takes over legal responsibilities for the waste from national waste management agencies is an issue discussed in Work Package 2 [40]. Further, the point at which employees of the ERO or its contractors take over practical charge from national waste management agencies is not yet defined – it is possible that shared responsibility may be applied to one or more of the steps indicated in section 4.6. Regardless of this, the responsibility of demonstrating compliance with the CFA/WAC will remain the responsibility of the national waste management agencies, and the ERO will verify the compliance of waste and waste packages with the safety and quality requirements.

5.3 Transport safety

Safe transport of radioactive waste and spent nuclear fuel is important to the viability of both national and shared waste management systems. There is extensive and positive experience of the safe transport of radioactive waste and nuclear fuel in many countries and internationally as described later in this section. Nevertheless, it is an area of high concern to the public, since it necessarily occurs across the public domain and affects communities that may feel they gain no benefit from the activity and have little influence on the decisions concerning such transport. The situation becomes more acute in the case of a shared system owing to the greater transport distances, trans-border aspects and further distancing of affected communities from any perceived benefit in such transports occurring.

A key element of the safe transport of radioactive waste is containment, and containment systems can be designed to satisfy requirements of transport and storage, and in some cases disposal. Thus, this section discusses the safety of containers that may be used for transport and for storage.

5.3.1 Safety guidance

The IAEA Regulations for the Safe Transport of Radioactive Material [24] establish standards of safety to provide an acceptable level of control of the radiation, criticality and thermal hazards to persons, property and the environment that are associated with the transport of radioactive material. The objective is to protect persons, property and the environment from the effects of radiation during the transport of radioactive material. This protection is achieved by requiring:

(a) containment of the radioactive contents;

(b) control of external radiation levels;

(c) prevention of criticality; and

(d) prevention of damage caused by heat.

The IAEA Regulations are supplemented by a hierarchy of Safety Guides including a general advisory guide [41], and guides related to emergency response to transport accidents [42], compliance assurance [43] and quality assurance [44] for the safe transport of radioactive material.

The IAEA requirements are satisfied firstly by applying a graded approach to contents limits for packages and conveyances and to performance standards applied to package designs depending upon the hazard of the radioactive contents. Secondly, they are satisfied by imposing requirements on the design and operation of packages and on the maintenance of packages, including a consideration of the nature of the radioactive contents. Finally, they are satisfied by requiring administrative controls including, where appropriate, approval by competent authorities.

Packages for transport of SF, HLW and ILW will be of the type designated “Type B” by IAEA. These packages are used to transport relatively large quantities of radioactive material. They must be able to withstand a wide range of accidents conditions and to satisfy tests including underwater immersion, fire impact and puncture.

5.3.2 Containment for transport and storage

Spent fuel

A key issue concerning spent fuel is the storage condition prior to acceptance by the shared waste management system. Fuel discharged from nuclear reactors is stored first under water in cooling ponds at the reactor site. After initial cooling it may be transferred to “dry store” vaults or casks for further cooling and storage, or it may remain in “wet store” in cooling ponds. If needed, fuel may be transported between facilities, e.g. for reprocessing (as in the UK and France) or to a centralised storage facility (such as the CLAB in Sweden). In this case, the fuel is loaded into re-usable transport containers under water and transported “wet” within sealed transport containers, see Figure 5.1.

Spent fuel accepted for disposal within the shared repository, would necessarily have been in national storage for 40 to 50 years, and possibly longer, following discharge for the nuclear reactor to allow cooling sufficient to meet repository thermal design requirements, see section 5.5. In this case, the SF will be suitable for dry storage, if not already in dry storage, and requirements on the shared waste management system can be simplified by only accepting “dry” spent fuel. Thus, pond storage and drying arrangements would not be needed at the encapsulation facility, which avoids the production of secondary radioactive wastes associated with pond storage.

Three general types of containers are possible for SF, e.g. see [45]:

- single purpose, transport only containers;
- dual purpose, transport and storage containers;
- multi (triple) purpose, transport, storage and disposal containers.

Transport only containers are designed to be re-usable, and only for relatively short-term use during the transport between facilities, after which they are cleaned, checked and returned for re-use as needed. Dual-purpose transport and storage containers are designed both to meet transport requirements and to provide safe storage of up to several decades if needed. They may be re-used but storage is part of their prime function. Multi-purpose containers (MPCs) would be disposed in the repository.

A common type of dual-purpose container is CASTOR[5] transport and interim storage cask, see Figure 5.2, which are widely used in Europe and elsewhere. The containers are manufactured from ductile cast iron and have been manufactured in various sizes and internal designs to accommodate different types of SF and HLW.

In response to the delays in disposal, and consequent demand for spent fuel storage, a number of concrete container types have been developed as an economic alternative to metal container options. Concrete cask systems typically consist of an inner sealed metal container housed within a ventilated outer concrete cask that protects the inner container.

An interesting possibility is the development of depleted uranium dioxide (DUO_2)-steel cermet (composite ceramic and metallic) materials. These have advantages over steel, providing provide better gamma shielding because of their higher density, and better neutron attenuation because of the oxygen content of DUO_2, which moderates neutrons [46].

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[5] CASTOR is a registered trademark of the Siempelkamp GmBh manufacturing company, but the name is commonly used to describe the generic concept of cast steel container for SF/HLW.

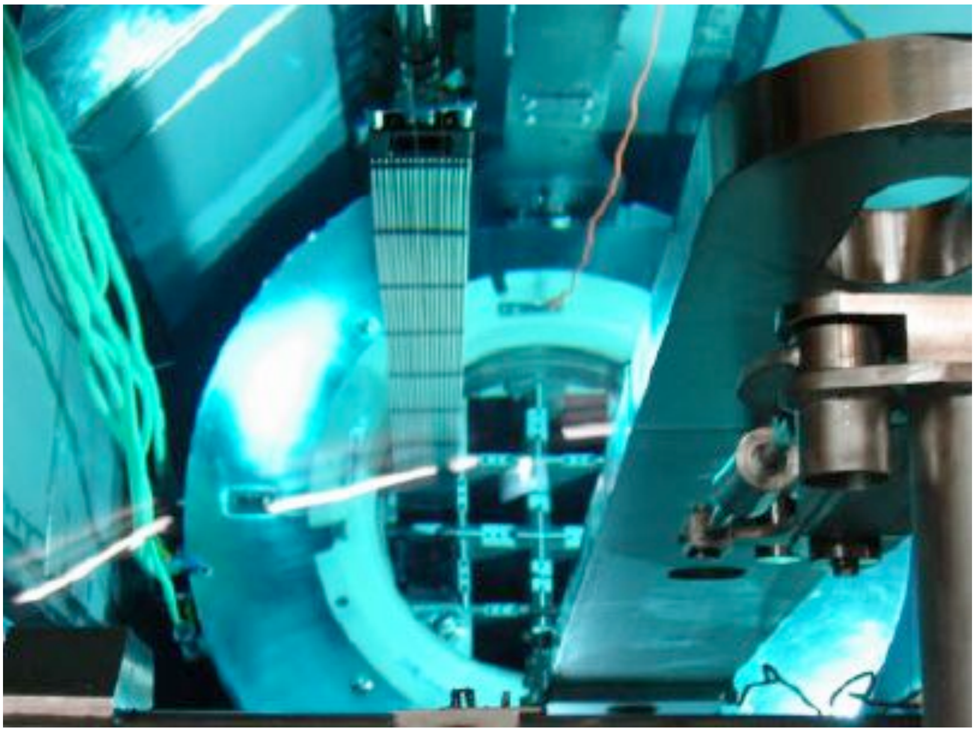

Figure 5.1: Loading a spent fuel in transport container (type TN13/2) under water

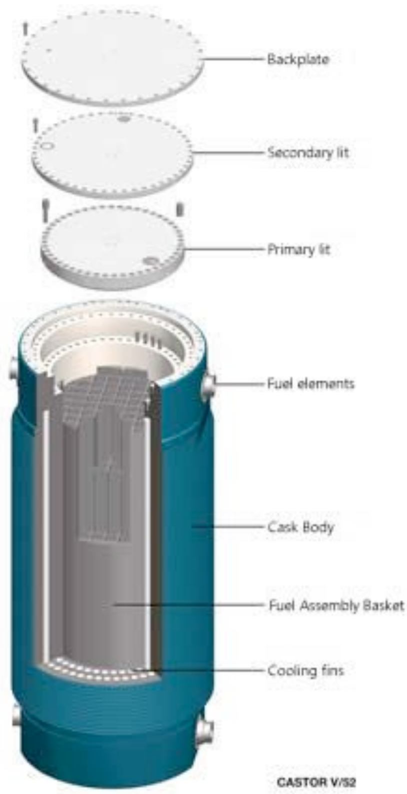

Figure 5.2: Construction of a CASTOR (V/52) transport and storage container

The main advantage of dual- or multi-purpose technologies is the reduction of the need to handle fuel assemblies for transfer operations between the different steps of spent fuel management, which implies [45]:

- reducing the need for handling unshielded fuel assemblies and thus associated dose and possibility of human error;
- minimising the need for transfer facilities and associated safety risks and costs;
- facilitating operations involved in the interface operations between different steps of the spent fuel management down to disposal, including safeguards inspection.

On the other hand, depending on repository concept, requirements for disposal may be more specific and potentially demanding than for storage/transport. In particular, it must be possible to open and unload a storage container, while a disposal container should be fully sealed to provide long-term containment in repository conditions.

Both single and dual purpose containers are in common use at present; multi purpose types are not favoured at present because of the high cost of a single package design to simultaneously meet transport, storage and disposal requirements, and because such a design could constrain possibilities for disposal. Rather, at present it is considered better to design a separate disposal container best suited to a given repository concept, once the repository concept is established. Relative merits of MPCs with respect to disposal are discussed further in section 5.5.

High-level waste

High-level waste for disposal is currently in the form of borosilicate glass (vitrified HLW) within stainless steel containers, such as produced by commercial reprocessing in France and the UK.

The HLW would be loaded into either transport only, or dual-purpose (transport/storage) containers, much as for SF. Figures 5.3 and 5.4 show the construction and appearance of a typical HLW transport only container.

Intermediate level waste

Intermediate level wastes arise in much more varied forms and packages. Most operational waste is packaged in 200 or 500 litre drums, but may be overpacked for transport, storage and/or disposal.

UK Nirex Limited[6] focused much of their work on waste package specification and conditions for acceptance of ILW and LLW waste, recognising that assurance of waste characteristics and standard packaging would be crucial to safety and to practicalities of transport and package handling, e.g. [47]. Figure 5.5 shows standard ILW packages designed by Nirex. The stillage containing 4x500 litre stainless steel drums is the predominant type for ILW in the UK; these would be placed in a re-usable shielded transport container (RSTC) for transport, see Figure 5.6. The other containers shown incorporate shielding for safe transport and operations as needed, which may depend on their content.

[6] The organisation formerly responsible for developing concepts for geological disposal in the UK; its functions have now been taken over by the Nuclear Decommissioning Authority (NDA).

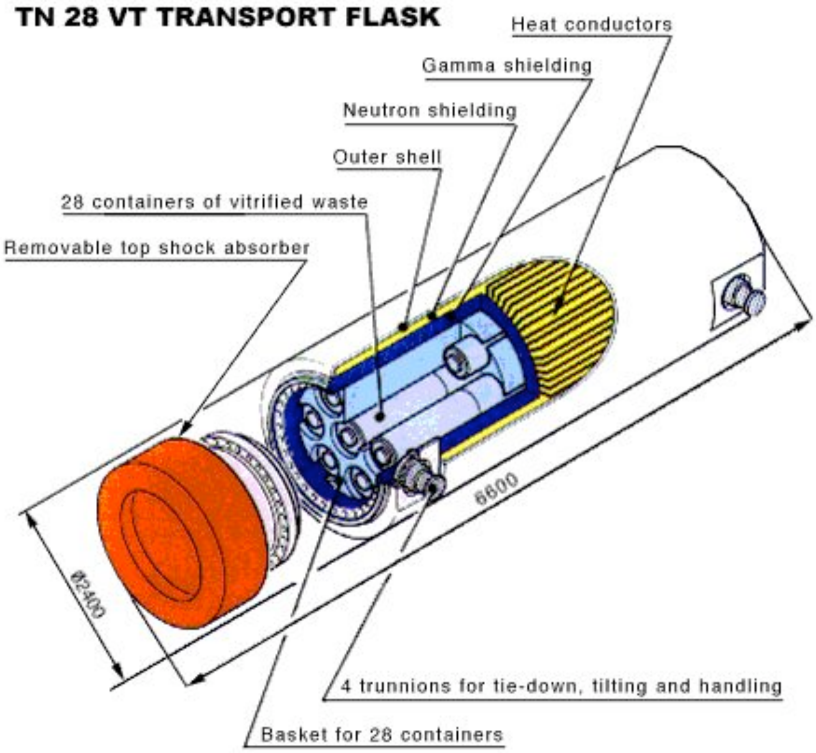

Figure 5.3: Construction of a Transnucleaire (TN 28 VT) HLW transport container

Figure 5.4: HLW transport container during shipping

Figure 5.5: Standard ILW packages as specified in the UK (not to scale), from [48]

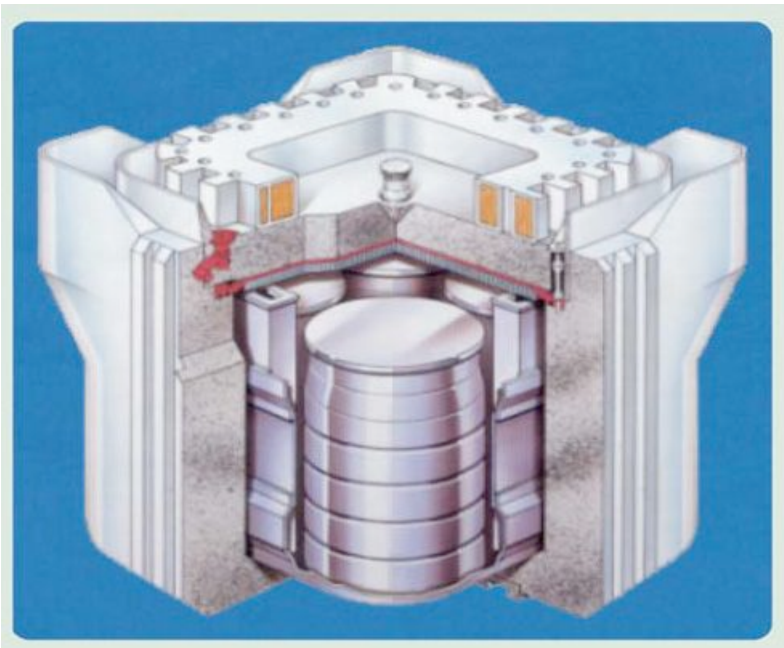

Figure 5.6: 4 x 500 litre ILW drums in re-usable transport container

5.3.3 Safety analysis and testing

Packages for transport of SF, HLW and ILW must be able to withstand a wide range of accidents conditions and to satisfy tests including underwater immersion, fire impact and puncture, see Figure 5.7. The tests are specified by the regulatory authorities based on worst conceivable accidents that could occur in transport, and adequate performance must be demonstrated before a licence will be granted for a given container or package type.

Numerous analytic studies and field tests have been done on the safety of transporting spent fuel and other radioactive waste by manufacturers of shipping containers as part of the licensing process, as well as by waste management agencies, national laboratories and private contractors. These include analysis of package performance under normal and accident scenarios, and dose assessments considering transport by various modes in both urban and rural areas [49].

Field tests have also been performed to subject containers to severe accidents. In the USA, the DOE sponsored a series of crash tests in the mid-1970s including: (1) a flatbed truck loaded with a full-scale cask crashed into a 700-ton concrete wall at 130 kilometres an hour; (2) a cask was broadsided by a 120-ton locomotive travelling at 130 kilometres per hour; and (3) a transport container dropped 200 metres onto soil as hard as concrete (the container was travelling 380 kilometres per hour at impact) [50].

In the UK in the 1980s, the CEGB conducted a live television demonstration of the integrity of a rail transport container for spent fuel. The test involved ramming an unmanned locomotive at 160 kilometres per hour into a container used for shipping spent fuel from the UK power stations for reprocessing.

More recently in Germany, a test was conducted of the impact of an exploding LPG[7] rail tank car onto a CASTOR spent fuel cask [51]. About 17 min after fire ignition the propane tank ruptured. This resulted in an expanding fireball, heat radiation, explosion overpressure, and tank fragments projected towards the cask, imposing severe mechanical and thermal impacts directly onto the cask, moving it 7 m from its original position, see Figure 5.8.

In none of these tests was a container damaged to the point that radioactive material would have been released. Post-test investigations of the containers demonstrated that no loss of leak-tightness or containment and shielding integrity occurred. The test results also indicated that analytical and scale-modelling techniques could predict vehicular and container damage in extremely severe accidents with reasonable accuracy. Thus the tests confirmed that spent fuel containers are capable of surviving very severe accidents and continuing to provide their nuclear safety functions.

5.3.4 Experience of transport safety

The worldwide experience of storing, handling, and shipping spent nuclear fuel and high-level waste is based on more than 50 years of operating nuclear reactors. Thirty thousand to 50,000 containers have been shipped by all surface modes of transport (i.e., road, rail, and sea) involving an estimated 100,000 tHM [8] [45; 49].

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7 Liquid Propane Gas

8 tHM = metric tonnes of heavy metal

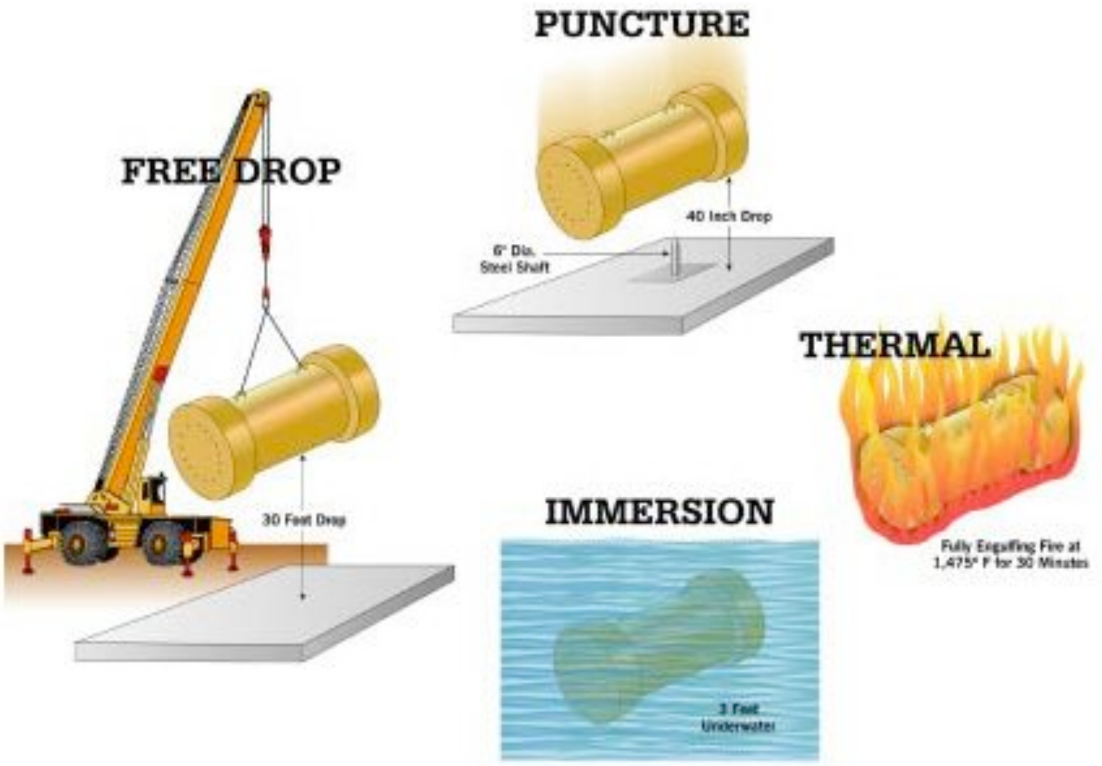

Figure 5.7: Standard test for SF containers specified by the USNRC

Figure 5.8: Post-explosion view of test of the impact of LPG container explosion (centre right) on a CASTOR SF container (left), from [51]

A case of a large operation for spent fuel transport is anticipated for the Yucca Mountain repository. A total amount of 70 000 tHM will have to be transported from storage sites around the USA; that is almost three times the SF amount considered in the shared European repository "large inventory", see section 4.2.1, and the transport would be over similar and larger distances.

In the US, between 1964 and 1997, 829 tHM were shipped by road and 1,445 tHM by rail; a total of 3,025 shipments. Although there were many more shipments by road, the tonnage of rail shipments exceeded the tonnage of shipments by road by a factor of about 2 [49]. It is reported that that four highway shipments and four rail shipments were involved in accidents between 1971 and 1995, only one of which resulted in detectable damage to the cask. Although the driver was killed in the accident, radiation surveys at the scene indicated that the structural integrity of the cask was not compromised, and there was no release of radioactive contents. No non-routine exposures to radioactive material have resulted from transport accidents.

The evidence showing the safety of the management and transport of spent nuclear fuel is impressive [49]. This reflects both strict standards for shipping containers (e.g. resistance to impact, fire, and water-immersion), and also the very stable form of the spent fuel. Unlike most hazardous materials, spent nuclear fuel is not a gas, liquid, or powder. In addition, neither mechanical nor thermal energy is present to serve as a dispersion mechanism in the event that the containers are penetrated or engulfed in fire. On the whole, undamaged fuel assemblies are very rugged and represent an effective containment barrier for most radionuclides.

5.3.5 Indicative doses and risks

Spent nuclear fuel

As noted above, no non-routine exposures to radioactive material have resulted from transport accidents involving spent fuel. There is no evidence that commercial nuclear waste transports has involved individual doses greater that those specified by regulators and waste management agencies, i.e. 20 mSv y^{-1} for registered radiation workers, 5 mSv y^{-1} for other workers and 1 mSv y^{-1} for members of the public, see Table 5.2. Actual individual doses attributable to transport and associated handling operations are probably much lower.

Transport operations are routinely monitored, although this usually focuses on ensuring that package surface contamination and doses remain within regulatory and operational requirements. In response to a stakeholder challenge, monitoring and dose assessments were made of the transfer of six spent fuel assembly transports from a nuclear power plant in Germany using CASTOR-V19 casks[9] in 1998-9 [52]. The findings were as follows:

- the collective dose during the loading of the CASTOR casks amounted to 4.5 mSv (gamma and neutrons) per cask at the most, and that the maximum individual dose amounted to 0.26 mSv;
- the collective dose during cask handling and transport amounted to 0.35 mSv (gamma and neutrons); the collective dose to the police escort was <0.1 mSv (gamma and neutrons).

[9] The CASTOR-V19 casks have capacity for spent fuel up to 10.2 tHM.

An assessment of radiological and conventional public safety of nuclear waste transport was made by Tunaboylu et al. as part of the Pangea project [53]; the radiological impact assessment is summarised below.

For incident-free transportation, radiation exposure to the public could result from external radiation in the vicinity of the transport containers (restricted to 0.1 mSv/h at 2 m) to people living along the transport route, people in vehicles sharing the transportation route and the public located near to transport stops. The estimated radiological impacts of to members of the public from both road and rail transport were negligible. The effective dose for a person standing 10 m from a transport travelling at 20 km/h was estimated at 0.025 microSv. This corresponds to an annual risk (of latent cancer fatality) to an individual less than 2 x 10^{-9}.

Assessments were also made for a range of accident conditions based on an accident classification system developed in relation to the Konrad facility in Germany. Table 5.3 summarises the radiological consequences of an accident resulting in a release of radioactivity during land transport (road or rail), dependent on the distance of a person from the site of the accident. The radiological consequences (given as the effective dose in mSv in the first year after the accident) were calculated for the transport of spent fuel in a type B (U) package (CASTOR type cask) in a study performed for the Swiss utilities. This study takes into account the maximum radioactivity release scenario, double seal failures of the seals on both closures and the most conservative atmospheric conditions. It thus represents a worst-case scenario, with an extremely low possibility of occurrence. Therefore, it can be considered as a bounding case for the estimation of the radiological risks of land transport, independent of the actual geographical location.

Severity category	BK 4 (single seal failure) Leakage rate ≤10^{-4} mbar l/s		BK 5 (double seal failure) Leakage rate ≤1 mbar l/s	
Probability of radioactivity release	PRR(road) = 1.3 x10^{-7} km^{-1}		PRR(road) = 2.8 x10^{-9} km^{-1}	
	PRR(rail) = 1.1 x10^{-8} km^{-1}		PRR(rail) = 2.3 x10^{-10} km^{-1}	
Distance (m)	Effective dose in the first year after accident (mSv)		Effective dose in the first year after accident (mSv)	
	Adult	Child	Adult	Child
100	1.6 x10^{-3}	2.1 x10^{-3}	15.6 (2.4)	20.6 (0.8)
200	7.0 x10^{-4}	9.4 x10^{-4}	6.8 (0.9)	9.2 (0.3)
500	2.5 x10^{-4}	3.4 x10^{-4}	2.4 (0.3)	3.3 (0.1)

( ) The figures in brackets represent the more realistic doses to the public based on inhalation doses [10].

Table 5.3: The probabilities of releases as a result of different severity categories of land transport accidents with spent fuel casks (CASTOR type) and their radiological consequences (effective doses in mSv), from [53]

Combining the probability of accident per kilometre with the calculated effective doses and a detriment-adjusted risk coefficient of 0.06 per Sv [11], yields individual risks per kilometre of

10 If members of the public stay in the immediate vicinity of the accident for the whole time and no clean-up is performed, the dominant portion of the effective dose comes from radioactive particles deposited on the ground and occurs over a period of one year after the accident. In reality, especially in the BK5 case, people would be evacuated from the area, which will be checked for radioactivity and remedial action taken if needed. In this case, the inhalation portion of dose received shortly after the accident is a more realistic estimate of the dose.

transport. These risks are dominated by the BK5 (double seal failure) accident, the higher calculated doses outweighing the lower probability. Taking, as the most representative case, an adult at 200m from the accident and exposure in the period following the accident mainly from inhalation, yields risks per kilometre of 1.5 $x10^{-13}$ for road transport and 1.2 $x10^{-14}$ for rail transport.

Applying the above assessment results to the case of the shared European disposal system:

- if a total inventory of 25,600 tHM of spent fuel (section 4.2.1),
- is transported in CASTOR type casks typically containing between 5 and 6 tHM so that 5000 cask shipments are required,
- where each is transported over distances of about 1000 km mainly by rail (section 4.4),

then the per kilometre probabilities of accidents leading to radioactive release given in Table 5.1 imply only a one in twenty chance of one BK4 type accident and a one in one thousand chance of a BK5 type accident during the entire transport campaign.

Radiological impacts from a BK4 type accident are entirely trivial, see Table 5.3; radiological impacts from a BK5 type accident are similar to the range of annual exposures to natural background and we see from the above that such an accident is very unlikely to occur.

This, of course does not mean that transport accidents will not happen, rather that any radiological consequences are very low to zero. Conventional risks from transport, accidents and from exhaust emissions will need to be considered in a full assessment of non-radiological impacts at a later stage.

High-level waste

Impacts from transport of HLW are liable to be lower than those related to spent fuel. The handling and containment systems for transport are similar and designed to meet the same radiological standard, thus the doses from routine situations are likely to be similar or less due to lesser inventory. The vitrified waste form is even more resistant to dispersal by impact or fire than spent nuclear fuel and does not contain the full spectrum of radionuclides, e.g. plutonium isotopes and volatile radionuclides; thus, in the event of an accident that breached containment, potential impacts would be much lower.

Intermediate-level waste

A comprehensive assessment of the radiological impacts from transport of ILW has been made in the UK considering reference case of transporting 168,000 m^3 of packaged waste from nuclear sites throughout the UK to a central repository location [54]; that is more than five times the volume consider in the shared European “large inventory” case (section 4.2.1).

Worst-case doses to hypothetical members of the public were calculated using conservative assumptions about their location, the number of packages to which they could be exposed and the package dose rates. In the worst-case case, an individual could receive a calculated dose of $2.5x10^{-3}$ mSv y^{-1}, that is one order of magnitude less that the Nirex Design Target of 0.02 mSv y^{-1} for members of the public. Actual doses are likely to be even

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[11] The most recent recommendations of the ICRP (Publication 103, 2008) propose a detriment-adjusted risk coefficient of 0.057 per Sv, which may be rounded to 0.06 per Sv. The risk coefficient for fatal cancer for the whole population is given as 0.05 per Sv.

lower. Maximum individual doses to workers were estimated at always less than the 2 mSv y^{-1} design target set by Nirex.

Upper and lower bounds of collective doses were calculated for incident free transport for a number of total volume and transport cases. The most representative case (maximum use of rail for transport of 168,000 m^3 of packaged waste) yielded collective doses to the public of between 9 and 21 man-mSv, and collective doses to the workers of between 11 and 16 man-mSv; these are collective doses over the whole transport campaign.

5.4 Operational safety of facilities

The main facilities comprising a shared European disposal system are the encapsulation facility for SF and HLW, and possibly overpacking of some ILW packages, and the deep geological repository. The facilities may include short-term storage areas for waste awaiting encapsulation and for encapsulated SF/HLW and packaged ILW awaiting disposal. The simplest case is that the encapsulation plant is sited at or very close to the repository. As noted in section 5.3.2, pond storage facilities would not be needed at the encapsulation facility, which avoids the production of associated secondary radioactive wastes.

The safety requirements are largely common for all facility operations, i.e. interim stores, encapsulation plant and the repository in its operational phase, and these are therefore described together.

5.4.1 Safety guidance

As introduced in section 5.1.1, the radiological protection requirements and criteria for radioactive waste storage and encapsulation operations, and the for operational period of a geological disposal facility, are the same as for any licensed nuclear facility, and are established in the IAEA Basic Safety Standards [20]. Safety requirements for predisposal management of radioactive waste, which includes storage are given in the IAEA safety requirements [25]. More specific guidance for predisposal management of high-level radioactive waste is given in [28]. An important feature is that safety should be assessed for routine operations and also for possible accidents.

5.4.2 Safe operations

Spent fuel and high-level waste

The main operational steps that will be required have already been introduced in Figure 4.2 and section 4.6, wherein it is noted that the exact steps may vary depending on whether waste is received in transport only or dual-purpose containers and whether the encapsulation facility is co-sited with the final repository. Transfers between vehicles can be minimised if a dedicated railhead forms part of a co-sited facility.

In Sweden, the reference option is to site the encapsulation plant at the CLAB facility, which is the centralised store for Swedish spent fuel [55]; the encapsulated waste would then placed in re-usable overpacks for transport to the final repository. In Finland, the reference option is to site the encapsulation plant at the site of the final repository. In both cases, the aim is to minimise waste handing operations and transport in the public domain, within the constraints of the established national waste management systems.

Incoming waste containers would be received, monitored for surface contamination and any indication of damage, and removed from their transport vehicle or rail wagon. If dual-purpose containers are used this provides flexibility for easy storage of waste awaiting encapsulation; safety is assured by the container design and a simple secure warehouse type building will suffice for storage of the loaded containers. Most transport only containers can also be used for temporary storage, but the incentive may be to return them for re-use promptly so as to minimise the number of containers required.

For spent fuel and high-level waste the key steps are the removal of the SF or HLW from the transport/storage containers, transfer to a prepared disposal container, and closing and sealing of the disposal container. These steps would be carried out using remote handling equipment within a sealed environment. The steps are very similar to those routinely carried out at nuclear power stations of transferring fuel to transport or dry storage containers, and similar techniques and precautions will apply. In general, however, the radiological hazard will be less because SF or HLW will only be accepted for encapsulation when it meets the thermal requirements for disposal, see section 5.5, which implies several decades of radioactive decay and cooling. Special attention would, however, have to be paid to defective or damaged fuel assemblies, or assemblies that had deteriorated during storage.

A significant demand applies to the closure of the disposal containers. In general these are not designed to provide sufficient shielding for open handling during operations, but rather to satisfy post-closure safety requirements, see below. Therefore the sealing and confirmation of sealing of the container must be done remotely. Figure 5.9 illustrates schematically the steps of unloading of incoming transport containers, loading disposal containers, welding and inspection. An important principle illustrated in the figure is that the areas of each operation are separated so that contamination and effects of any anomaly at any one step can be contained.

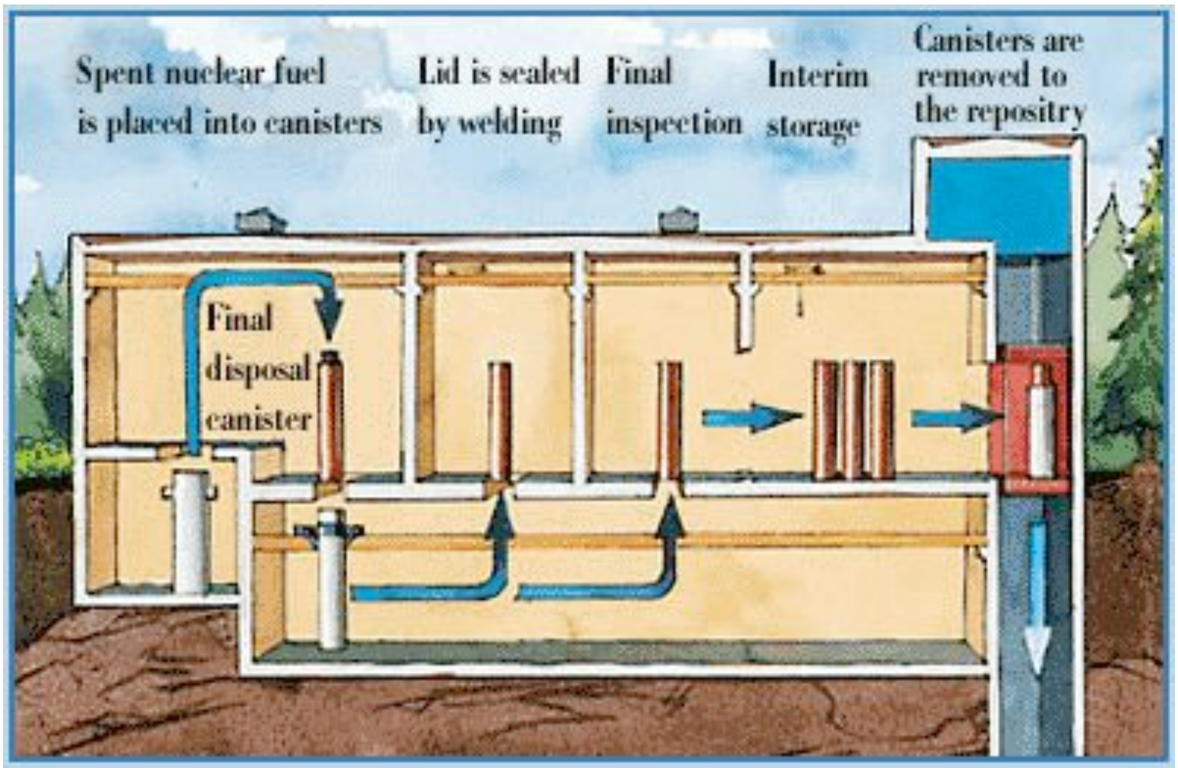


Figure 5.9: Schematic of encapsulation facility in the Finnish programme

The requirements for operational safety would be fully incorporated in to the design of the encapsulation facility and repository. This includes consideration of:

- designation of areas according to potential for external radiation and contamination;
- adequate shielding and contamination control;
- independent ventilation and filter systems, environment control and monitoring systems;
- provision for safe maintenance and replacement of equipment;
- minimisation of accident potentials, e.g. attention to heavy load movements, sources of ignition etc.
- emergency access and evacuation routes.

The Finnish regulatory agency, STUK, has issued guidance for the operational safety of a disposal facility for spent nuclear fuel that includes the spent fuel handling and encapsulation processes [56]. This includes technical design requirements in terms of:

- limitation of occupational exposure to radiation;
- limitation of radioactive releases;
- radiation monitoring;
- safety classification (of systems, structure and components);
- ensuring safety functions;
- prevention of criticality accidents;
- prevention of fire and explosion hazards;
- consideration of external events;
- control of nuclear materials;
- construction and operation of underground facilities.

For example, in relation to ensuring the safety functions, the guidance requires the functions at the 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. Safety systems that shall be ensured against single failure include:

- systems needed to prevent overheating of spent nuclear fuel bundles;
- radiation monitoring systems needed for accident follow-up and mitigation and the radiation monitoring system in the hot cell for handling of spent fuel elements;
- underpressurising and filtration systems in rooms into which large quantities of airborne radioactive substances may be released;
- monitoring systems for discharges of radioactive substances;
- fire alarm and extinguishing systems in areas where a fire could cause a significant release of radioactive substances within the facility or to the environment.

Further, the handling systems of spent fuel elements shall be designed so that a single equipment failure can not cause a drop accident or another kind of accident where spent fuel bundles could be severely damaged. The handling systems for spent fuel transport casks and waste canisters shall be designed so that a single equipment failure cannot cause a drop accident or another kind of accident where significant amounts of radioactive substances could be released from the cask or canister.

Intermediate level waste

A comprehensive assessment of the safety of a geological repository for intermediate and low level waste, considering above and below ground elements, is given in [57].

Issues are similar as described above for SF/HLW, although the potential for very high external dose rates as associated with unshielded waste are less. The variety of waste packages and different levels of external irradiation and contamination hazard may lead to more varied methods of handling and emplacement, see section 5.4.4.

5.4.3 Repository specific features

With respect to the repository itself, key decisions that impact on the management of underground radiological safety are:

- the disposal container design, especially the loaded weight and level of shielding;
- whether waste access is via an inclined drift (ramp) or vertical shaft.

Both of these decisions will depend on the geological and geotechnical factors at a selected site and also on more strategic decisions, e.g. the importance given to retrievability and relative importance given to different barriers in the long-term safety case. The following sections (5.4.4 and 5.4.5) discuss options with respect to disposal container and repository access.

5.4.4 Disposal containers and emplacement

Spent fuel and high-level waste

The well-known copper shell canister with steel insert, developed jointly by SKB and Posiva (KBS-3V concept), is designed to contain about two tonnes of spent fuel, see Figure 5.10. It has a length of approximately 4.8 m and a diameter of 1.05 m, and will have a loaded weight of about 25 to 27 metric tonnes. The 50mm thick shell is made of pure oxygen-free copper; there are two versions of the steel insert, one for BWR and one for PWR fuel assemblies. The decay heat of SF disposed in one canister is limited to 1,700 W, to ensure the temperature requirements of the buffer are met [58]. The oxygen-free copper shell is specified so as to provide very long containment, potentially in the range 10^5 to 10^7 years.

In Switzerland, the containers considered by Nagra in the Project Opalinus Clay are constructed of carbon steel and designed to provide complete containment for only 1000 years to cover the period of heat generation, although actual canister lifetimes of 10,000 years are reasonably expected. The reference design concept for SF canisters involves a cast steel body, with a machined central square channel fitted with crossplates to permit emplacement of either 4 PWR or 9 BWR fuel assemblies. The canisters are 1.05 m diameter, 4.6 m in length, have a wall thickness of 150 mm, and weigh ca. 26 metric tonnes when loaded [59].

Work in SAPIERR I (Appendix A in [4]) showed that similar design containers, of lengths between 3.7 m and 5.0 m and with alternative insert designs, could accommodate the foreseen spent fuel types from all the SAPIERR I countries including from fuel from VVER, CANDU and RBMK reactor types.

Figure 5.10: Copper-shell with steel insert disposal container parts

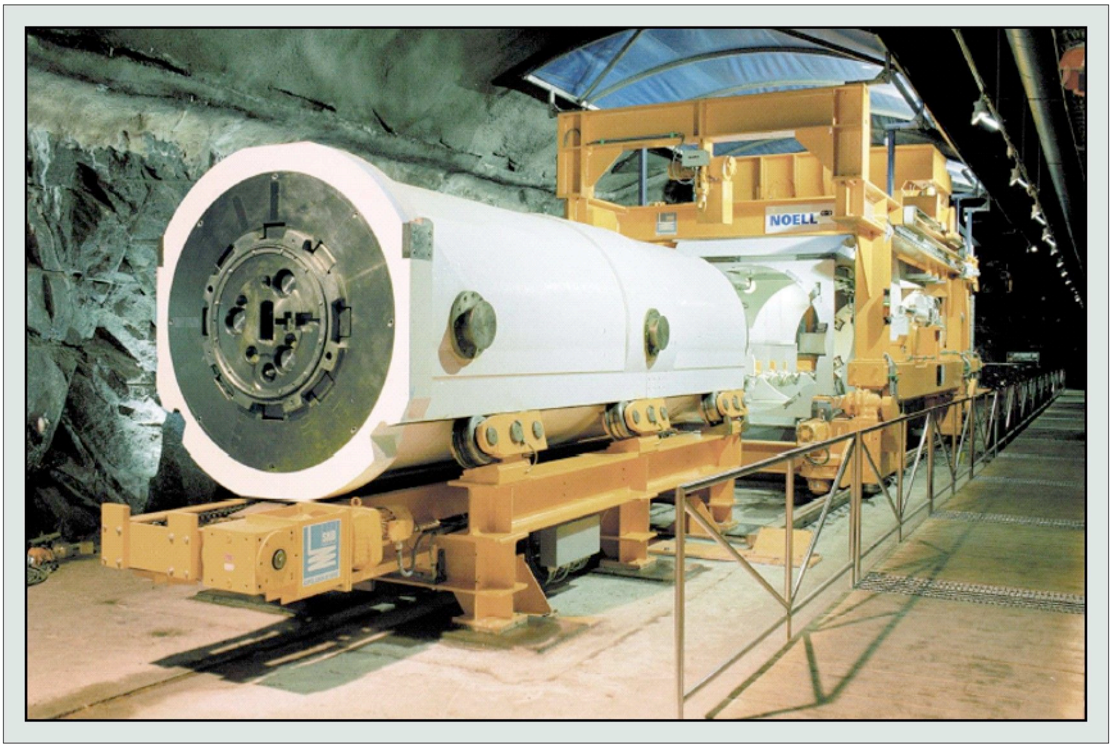


Figure 5.11: Prototype deposition machine at the Aspo Hard Rock laboratory

In the above cases the canisters are designed for post-closure performance and do not provide sufficient shielding for operational purposes. For the Swedish KBS3V canister, the design requirement is that the surface gamma-radiation dose does not exceed 1 Gy/h, which is designed to minimise the importance of radiololytic effects around the canister after emplacement. Dose rate calculations for a canister filled with 40-year old fuel indicated gamma and neutron dose rates of 350 and 20-40 mGy/h respectively. Further calculations showed gamma dose variation between 100 and 500 mGy/h depending on position on the canister surface (p. 86 in [60]). For comparison, container surface dose rates for SF/HLW transport are generally limited to be less than 0.2 mSv/h (0.2 mGy/h gamma).

In the KBS-3V concept, the emplacement process will consist of canisters being moved in shielded transport casks and then transferred to a remote-controlled and radiation-shielded deposition machine. The borehole will first be lined with rings of bentonite and then the waste package will be lowered into the borehole. When all the holes in a deposition tunnel are full, the tunnel will be backfilled with a mixture of bentonite clay and crushed rock [58]. Figure 5.11 shows the full-scale prototype canister deposition machine at the Aspo Hard Rock Laboratory in Sweden.

In the Swiss case, similar shielding and handling requirements would be required, but horizontal emplacement is envisaged. The waste package is transferred to the disposal tunnel in the shielded transport vehicle and then transferred to the deposition vehicle where it is positioned on a pedestal of compacted bentonite blocks supported by a steel frame. This assembly is moved into the disposal position and set down, after which the deposition vehicle withdraws. The granular bentonite buffer material is then filled around the waste package using a conveyor or pneumatic system. The whole procedure is carried out remotely, monitored by cameras and other sensing equipment mounted on the vehicles [59].

An alternative is to use a disposal container that also provides shielding such that some degree of worker proximity is feasible, which may reduce difficulties issues associated with remote emplacement and backfilling. The reference concept developed in Germany for the emplacement of SF at Gorleben uses massive, self-shielded containers (POLLUX casks). These would be emplaced axially on the flat floor of unlined drifts excavated in the salt host rock, which are then. backfilled with crushed salt. In this concept, disassembled SF is placed into POLLUX casks (1.5 m diameter, 5.5 m length), which are welded shut. The inner part of the POLLUX container is made of stainless steel, while the thick outer wall is made from nodular cast iron with two rows of axial boreholes that contain neutron-moderating material. A loaded POLLUX cask weighs about 65 tonnes and has a surface dose rate of less than 0.2 mSv/h. Each cask will hold the fuel from up to ten disassembled PWR assemblies, i.e. about 5 tHM.

This concept has been extensively tested and is regarded as fully feasible [61]. It has been suggested, however, it is less than optimal mainly in respect that the costly POLLUX casks serve little function in the post-closure period. Two alternatives are therefore also being investigated – the BSK3 and DIREGT concepts. The BSK3 concept envisages transfer of the fuel from the CASTOR casks, see section 5.3, in which SF is presently stored, to BSK3 containers, which are thin-walled (50 mm) steel containers. These would require shielding during transfer and emplacement and would be placed in vertical boreholes drilled from tunnels. Preliminary studies show that the costs for repository operations can be reduced significantly and costs for disposal casks reduced by 50%. A second alternative is to dispose directly of the CASTOR casks. The key feature here is one of avoiding the need for a step of transfer of spent fuel from the CASTOR casks to disposal casks. Although this means a high cost container is disposed, it can reduce worker doses and costs of fuel transfer facility operation and maintenance; importantly the transfer step, which is a potential bottleneck in the disposal operations, is avoided [61].

The Japanese CARE concept also considers the direct use of large steel or concrete multipurpose (transport-storage-disposal) containers. In this concept, large, steel multi-purpose containers (MPCs) or concrete disposal casks, either of which can hold up to approximately 20 HLW flasks or multiple fuel assemblies, are emplaced upright in large ventilated caverns for between 100 and 300 years to allow cooling and inspection. At the end of this period, the caverns can be backfilled with bentonite-based or cement-based backfill to complete the disposal. The advantage of this concept is mainly that a smaller repository footprint is achievable since disposal is not finalised until after a substantial reduction of heat output from the waste; the concept also offers longer-term easy retrievability.

A further general trend is the development of the integrated package or "supercontainer" disposal concept. The idea evolved from the recognition that emplacement of buffer materials around a waste canister could be problematic in underground conditions. The idea, therefore, of an integrated package, in which the compressed bentonite is held in place around the waste container within an outer overpack, and which would be emplaced as a single unit was suggested in the late 1990s. This may give increased confidence in the quality of buffer as the packages could be assembled above ground or in a below ground preparation area where quality controls could be better applied [33].

SKB and Posiva are currently developing and testing an alternative deposition concept for SF of emplacement in small horizontal tunnels (the KBS-3H concept). The KBS-3H "super-container" consists of a copper SF container loaded within prefabricated high-density bentonite blocks, retained by a steel perforated cylinder and end plates, see Figure 5.12. The perforated sleeve allows wetting of the bentonite immediately following emplacement in the disposal tunnel. Aspects of this concept, including full-scale handling and emplacement will be tested at the Äspö Hard Rock Laboratory [62].

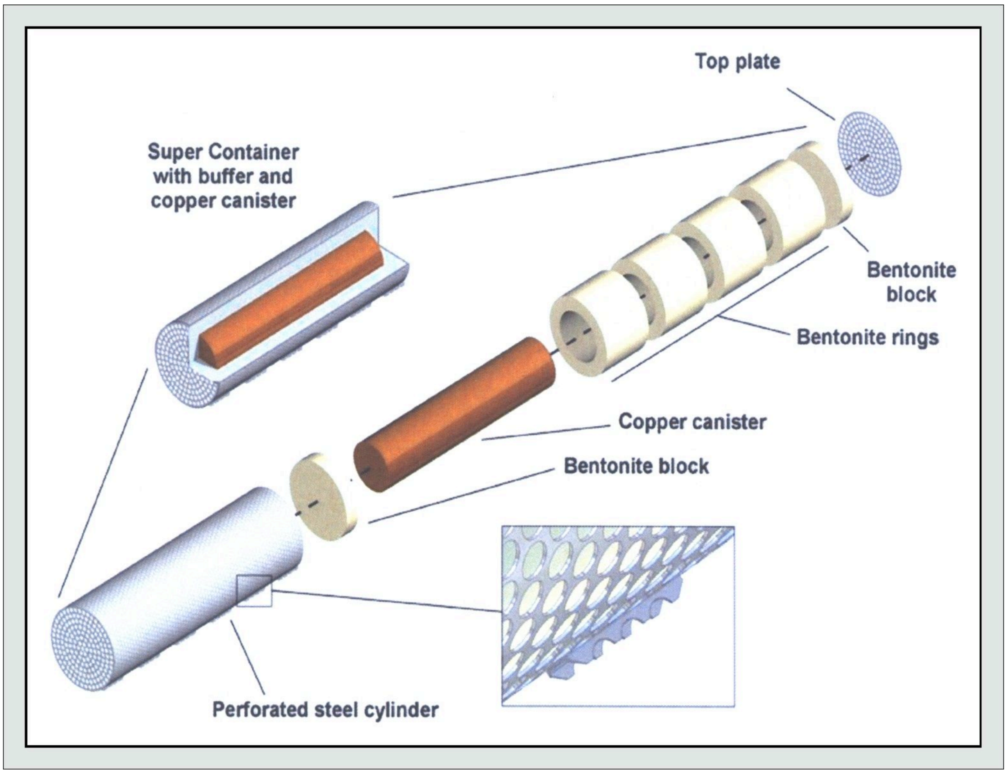


Figure 5.12: KBS-3H "supercontainer" conceptual diagram

In Belgium, Ondraf/Niras are also examining the potential of a ”supercontainer” designs for HLW and SF. The current Belgian design consists of a relatively thin (30 mm) carbon steel overpack, surrounded by a thick (ca. 700 mm) high-alkalinity concrete buffer within a thin (6 mm) steel envelope. Each HLW supercontainer will contain two stainless steel canisters of vitrified waste and will be 2.0 m in diameter, approximately 4.0 m long and weigh 30 tonnes. Supercontainers containing four UOX spent fuel assemblies will have a diameter of approximately 2.1 m, a length of 6.1 m, and weigh a maximum of 60 tonnes, whilst for MOX supercontainer will contain only one fuel assembly spent fuel the same measurements will be approximately 1.6 m, 5.2 m and 31 tonnes [63]. The concrete provides radiation shielding during transfer and underground emplacement, and promotes the life of the steel overpack by providing an alkaline environment that minimises corrosion.

Thus, in summary, a variety of disposal container options have been developed and tested where for SF/HLW the main choices are between:

- disposal only containers designed for post-closure performance and requiring shielding during underground operations, with loaded weight typically in the range 20-30 tonnes;
- using multipurpose (transport-storage-disposal) containers that incorporate shielding needed during underground operations, with loaded weight typically in the range 60-70 tonnes, and containing larger amounts of SF than the disposal only types;
- integrated disposal package (supercontainer) types, which may or may not provide the full amount of shielding needed during underground operations, with loaded weights in the range 30-60 tonnes.

Moving loads of this magnitude is not unusual in industrial applications, although specific issues arise in the underground and where shielding or remote handling is required. Waste canister transfer and emplacement technology, and heavy load emplacement technology, are among the topics currently being investigated within the EC project: Engineering Studies and Demonstration of Repository Designs (ESDRED) [64]. The overall objective of ESDRED is to demonstrate the technical feasibility at an industrial scale of activities needed to construct, operate and close a deep geological repository in compliance with requirements on operational safety, retrievability and monitoring.

Intermediate level waste

Equivalent issues arise as for SF/HLW although on less extreme scale.

For example, in the UK reference repository concept for ILW/LLW [48] most ILW waste will be received at the repository as unshielded waste (UILW). This will be taken underground in the re-usable shielded transport containers (RSTCs) and transferred in a shielded facility to a rail transport system (separate from worker access and other transfer paths) that will take the waste to the disposal caverns where it will be remotely emplaced by overhead crane. On the other hand, LLW and ILW in disposal packages that include necessary shielding, termed shielded ILW (SILW), will be transported underground and emplaced in caverns of simpler design by stacker truck.

The reason for the relatively elaborate treatment of UILW is economic in view of the relatively large volumes of such waste for disposal in the UK. On the other hand, in the Swiss reference concept for long-lived L/ILW [59] all waste will be placed and grouted in large concrete containers above ground, which will provide shielding for underground transport and emplacement.

5.4.5 Repository access

A key decision is whether waste access will be by shaft or inclined tunnel. Shafts are less expensive in construction and are likely to be used to first reach the potential repository horizon, e.g. for RCF development and construction. An inclined tunnel offers advantages in terms of heavy load transfers required to carry waste packages from the surface to their emplacement position.

A repository might be developed following normal deep mining practice with the main access for construction and removal of excavated rock via vertical shafts and subsequently waste emplacement could also be by this route. The German reference concept for disposal in salt dome at the Gorleben envisages lowering of the 65 te POLLUX casks, see above, to the repository level, 870 m below ground, in a waste handling shaft by a shaft hoisting system capable of handling loads of up to 85 tonnes.

Safe transfer of heavy loads by shaft has been demonstrated in mining and is clearly feasible; double cabling and winding, and arrestor systems, can reduce the risk of uncontrolled falls to a very low level. Nevertheless, there is at least a perceived risk of serious accidents associated with transfer of heavy radioactive loads by vertical shafts. This provides an additional motivation for arranging waste access by inclined tunnel where this is feasible.

The development of an inclined tunnel access, or ramp, is proposed in several repository projects. The motivations for inclined tunnel access are to provide more convenient vehicular access in general and to avoid the transfer of heavy waste containers by vertical shaft. In particular, inclined tunnel access would allow shielded or self-shielding waste containers to be taken directly from the surface to near to the final emplacement position by a single vehicle, thus minimising the need for inter-vehicle transfers or reorientation of containers as might be required at the shaft head and foot. This improves the technical convenience of waste transport to the repository horizon and also minimises steps at which doses to workers might be implied.

Tunnel access is most convenient where it is possible to construct into a rising topography, e.g. as at Yucca Mountain, or where the repository depth is not excessive in low permeability rock, e.g. at the locations considered in Sweden and Finland (ca. 500 m below ground). It would not be favoured at great depth on cost grounds, or where the access tunnel would have to cross high permeability strata or other potentially water-bearing features at depth, which could pose a groundwater control problem (potential inflow of pressurised groundwater) and risk of flooding.

5.4.6 Indicative doses and risks

The regulatory guidance specifically for the operation of encapsulation and disposal facilities, e.g. [56] indicates that external dose rates and risk for internal exposure should both be low, and certainly within the statutory limits, i.e. 20 mSv per year as provided in the IAEA Basic Safety Standards and EC Directive 96/29, see sections 5.1.1 and 3.1.5.

Potential for doses arises first on receipt of waste packages. Data from the HABOG long-term storage facility in the Netherlands [65] indicates the following for a campaign of unloading of research reactor spent fuel from a MTR-2 cask. The highest doses arose from inspection of the closed transport container (up to 7 μSv). Once the containers are accepted, all handling is remote controlled and the dose is negligible. Maximum individual dose for one MTR-2 unloading and storing campaign was 10.4 μSv. The average dose to 3 operators and 2 radiation protection officers was about 8 μSv during the 8-day campaign. The collective dose is hence only about 40 μSv. That is much lower than the 4.5 mSv (4500 μSv) estimated for the loading of commercial fuel into CASTOR casks discussed in section 5.3.5.

Safety assessment of HLW repository operations by ANDRA, in France, indicate the highest individual doses to workers would be between 2 and 4 mSv/year [66]. Doses at this level would be associated with primary package reception operations, the transfer and emplacement of disposal packages in the cell, in addition to installation monitoring and maintenance operations. The values associated with other activities are lower than 2 mSv/year. These results, obtained in the framework of the evaluation of the feasibility of the repository, may change in the course of a later radiological protection optimisation approach.

ANDRA have also studied accident scenarios (for example, fire or a package being dropped). They do not lead to a risk of radioactive material being released because of the safety provisions proposed; these include the safety systems installed in the shafts, the limiting to a few metres of the height from which packages could fall during handling operations, the use of shielded transportation casks and the robust design of disposal packages.

For the public on the periphery of the site (at a distance of 500 m from surface installations), external exposure is nil on account of the distance from surface nuclear installations. The impact of some radioactive gases (mainly radon) emitted into the atmosphere in the environment has been estimated on the basis of pessimistic hypotheses. Pessimistic estimates give a dose (0.001 mSv/year), which would be negligible compared with regulatory limit (1mSv/year) and compared with natural radioactivity (on average, 2 mSv/year).

Safety assessment of L/ILW repository operations by Nirex, in the UK, indicate that doses to workers would be within the Nirex design target of 2 mSv/year or could be kept within the design targets by use of elimination, mitigation and protection techniques [57]. For the reference operating strategy of backfilling after 100 years, peak dose rates to members of the public for routine ground and stack releases are well below the statutory limit of 1 mSv/y and well within the 0.02 mSv/y Nirex Design Target.

In order to estimate collective doses, work force size and numbers involved in active operations are relevant. SKB has estimated that during the operational periods, the storage facility CLAB will require around 60 personnel, the encapsulation plant will require approximately 30, and the repository will require approximately 200 [67], although the dose distribution is not given. It is estimated that 25 to 30 persons will be employed underground during a Swiss repository operational period. In Germany, repository operations are estimated to employ about 100 staff [68].

5.5 Repository post-closure safety

5.5.1 Approach to repository post-closure safety

As introduced in section 2.2, a defining principle of the geological disposal concept is that, after repository closure, the disposed waste should remain safe and secure even without monitoring or further protective actions [8]. To ensure the prospective long-term safety of a geological repository, which is a complex natural and engineered system, is an unusual and demanding challenge. In the long term, as remarked in section 5.1.1, the source will not be under direct control and it cannot be assumed that a radiation protection programme will be in place at the time when radionuclide releases from the repository may occur.

Demonstration of safety therefore relies on demonstration of the feasibility and performance of the technological elements envisaged, and on modelling and assessment studies that seek to illustrate the long-term performance of the disposal system and show that this meets standards of protection that would be acceptable today. The aim is not to forecast the future performance of the disposal system; rather the aim is to show that, taking account of uncertainties, all likely evolutions of the system lead to acceptable performance and that any events that might have detrimental impacts on the system are of sufficiently low probability that the risk to humans and the environment is acceptably low. This is achieved by a choice of site and host rock that offer favourable qualities for long-term isolation and containment of the radioactive waste, and by the development of compatible engineered barriers that further protect and contain the radioactive waste in concert with the natural qualities of the chosen site and host rock. The arguments and evidence that describe, quantify and substantiate the safety, and the level of confidence in the safety, of the geological repository are assembled in a safety case [69].

Estimates of performance are most often expressed in terms of radiological doses and risk. It must be stressed, however, that as advised by the ICRP [23]:

"(41) … Doses and risks, as measures of health detriment, cannot be forecast with any certainty for periods beyond around several hundreds of years into the future. Instead, estimates of doses or risks for longer time periods can be made and compared with appropriate criteria in a test to give an indication of whether the repository is acceptable given current understanding of the disposal system. Such estimates must not be regarded as predictions of future health detriment." and

"(70) In a long-term radiological assessment, doses or risks are calculated under reasonable selected test conditions as if they were doses or risks … they should be considered as performance measures or 'safety indicators' indicating the level of radiological safety provided by the disposal system. … "

The facts that (1) post-closure safety assessments aim only to illustrate (not to predict) performance and safety, and (2) that the estimates of radiological doses and risk are 'safety indicators' not actual doses or risks, have important implications when considering comparisons between geological disposal options, see section 5.6.

5.5.2 Safety guidance

Requirements for promoting and ensuring the safety of geological disposal have been developed at an international level by the IAEA in a 'Safety Requirements' document [8]. Besides an objective and criteria, see section 5.1.1, the document sets out requirements for planning and for developing geological disposal facilities in 23 areas, see Box 5.2.

Box 5.2: IAEA safety requirements for geological disposal – topic headings, from [8]

Safety Requirements for planning geological disposal facilities

Legal and organizational framework

- Requirements for government responsibility
- Requirements for regulatory body responsibility
- Requirements for operator responsibility

Safety approach

- Requirements concerning the importance of safety in the development process
- Requirements concerning passive safety
- Requirements for an adequate understanding and for confidence in safety

Safety design principles

- Requirements for multiple safety functions
- Requirements concerning containment
- Requirements for isolation of the waste

Requirements for the development, operation and closure of geological disposal facilities

Framework for geological disposal

- Requirements for step by step development and evaluation

Safety case and safety assessments

- Requirements concerning preparation of the safety case and safety assessment
- Requirements on the scope of the safety case and safety assessment
- Requirements concerning documentation of the safety case and safety assessments

Steps in the development, operation and closure of geological disposal facilities

- Requirements on site characterization
- Requirements for geological disposal facility design
- Requirements for geological disposal facility construction
- Requirements for geological disposal facility operation
- Requirements for geological disposal facility closure

Assurance of safety and nuclear safeguards

- Requirements on waste acceptance
- Requirements concerning monitoring programmes
- Requirements concerning post-closure and institutional controls
- Requirements in respect of nuclear safeguards
- Requirements concerning management systems

An important and recurring feature of the IAEA requirements is emphasis on step-by-step development of plans, safety assessments, the safety case, development of a facility, and regulatory review and decision making. This is echoed by the NEA documents on building and communicating confidence in long-term safety [70] and the safety case [69] for of deep geological repositories.

The IAEA ‘safety requirements’ will be supported by guidance (in preparation) that further expands on how the safety requirements may be interpreted and implemented [71].

5.5.3 Geological disposal concept

The selection of geological disposal as a safe, ethical and environmentally sound option that is technically feasible has been confirmed and endorsed in numerous national studies and internationally over many years, e.g. [72, 73, 74, 75]. The conclusions concerning its feasibility and safety are built on extensive research and development programmes in many countries over several decades, in which of the order of 10 billion US dollars has been invested [76].

It is acknowledged that the development of a deep geological repository is a complex and challenging enterprise. Therefore, the development will be carried out in a number of steps, with opportunities for review of safety, technical procedures, environmental impacts and acceptability. This has been explicitly recognised or proposed in the repository programmes in most countries, e.g. [77, 78, 79]. It is also considered that a deep geological repository can be developed in a flexible and reversible manner. This will provide opportunities to assimilate new understanding and new technology, and to address previously unforeseen difficulties or changes in strategy that may arise in the course of a project. It could, if needed, involve retrieval of some or all of the emplaced waste [80, 81]. The phased and reversible approach to repository development may also contribute to social acceptability.

The geological disposal of radioactive wastes is based on the principle that a suitably chosen deep rock environment is stable and will be largely unaffected by environmental change for a million years or more. That is for a time period longer than since the appearance of modern humans in Africa about 100,000 years ago, and during which Northern Europe has been exposed to several ice ages. Materials that are placed in a well designed repository, deep underground in a suitable geology, will for practical purposes be permanently isolated from humans and the environment in which we live.

Geological repositories are based on the concept of multiple barriers that work together to provide containment and isolation of the waste, e.g. see Figure 5.13.

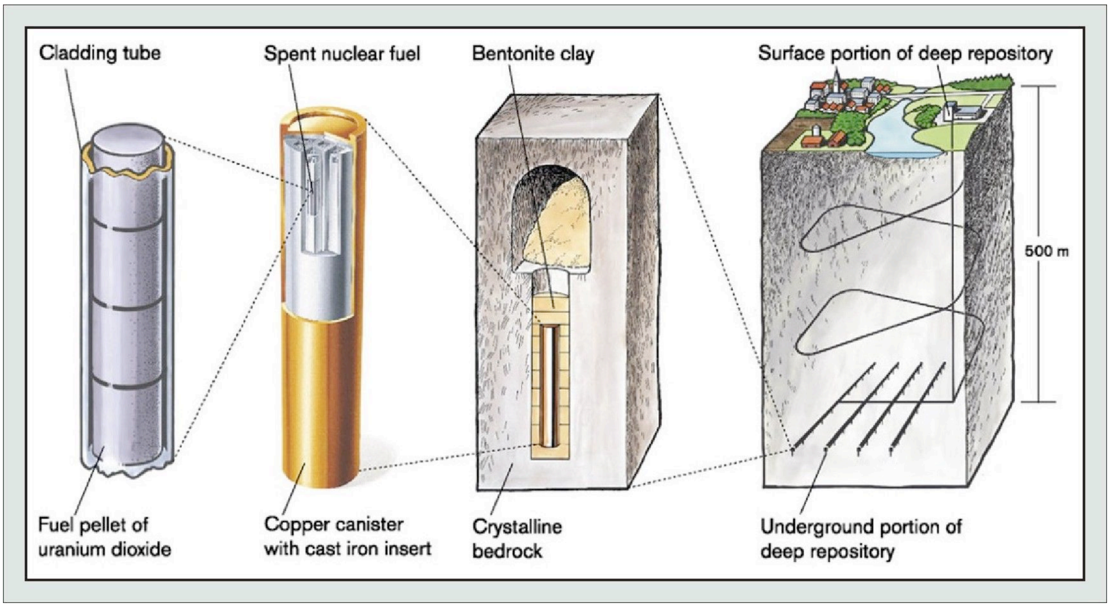


Fig. 5.13: Example of a multiple barrier repository system – the KBS-3V method developed for the disposal of spent fuel in Swedish bedrock

5.5.4 System barriers and safety functions

The system includes:

- engineered barriers – consisting of the waste form itself, physical containment and further physical and chemical barriers constructed as part of the repository, and

- natural barriers – provided by the host rock in which the repository is constructed and the surrounding geological environment.

The engineered barriers provide initial containment and protect the waste from contact by groundwaters that are present in most rock formations. They also provide longer-term physical and chemical protection, which limits the ingress of groundwater and the release of radionuclides even after some physical degradation has occurred. The natural geological barriers provide isolation and protect the engineered barriers so that they can operate as designed. They will also retard and disperse longer-lived radionuclides that may eventually be released from the engineered barriers.

The selection, design and function of barriers within a geological repository vary according to waste type, host rock and other factors. Usually, however, the following elements can be identified:

- a waste form that is stable, resistant to degradation and resistant to leaching by groundwater;

- containers that protect the waste form and prevent groundwater reaching it for at least several hundred years and, in some concepts, for tens or even a hundred thousand years. By this time, most activity will have decayed inside the container;

- buffer material that protects the containers, preventing or limiting water flow around them and absorbing any mechanical disturbance that might be caused by host rock movements (e.g. associated with major earthquakes). It may also sorb or retain radionuclides that eventually escape from the container;

- backfill material and seals (placed as the repository is closed), that prevent water or gas movement along underground tunnels and passages of the repository, so that these cannot provide a more rapid path for radionuclide movement;

- the host rock and immediate environment of the repository that provide stable (or only very slowly changing) mechanical, chemical and water flow conditions around the engineered barriers for very long times, allowing them to operate as designed and contain radionuclides for much longer than if they were constructed at the ground surface;

- the other rock layers and units, soils and waters around and above the repository that slow down, immobilise, dilute and disperse releases of radionuclides from the engineered barriers and host rock, so that the eventual release of radionuclides to the biosphere is only expected at long times in the future and at very low levels, so that any hazard is negligible.

Figure 5.14 shows an example of a system of multiple barriers and their expected safety functions, in this case from the Swiss programme [59].

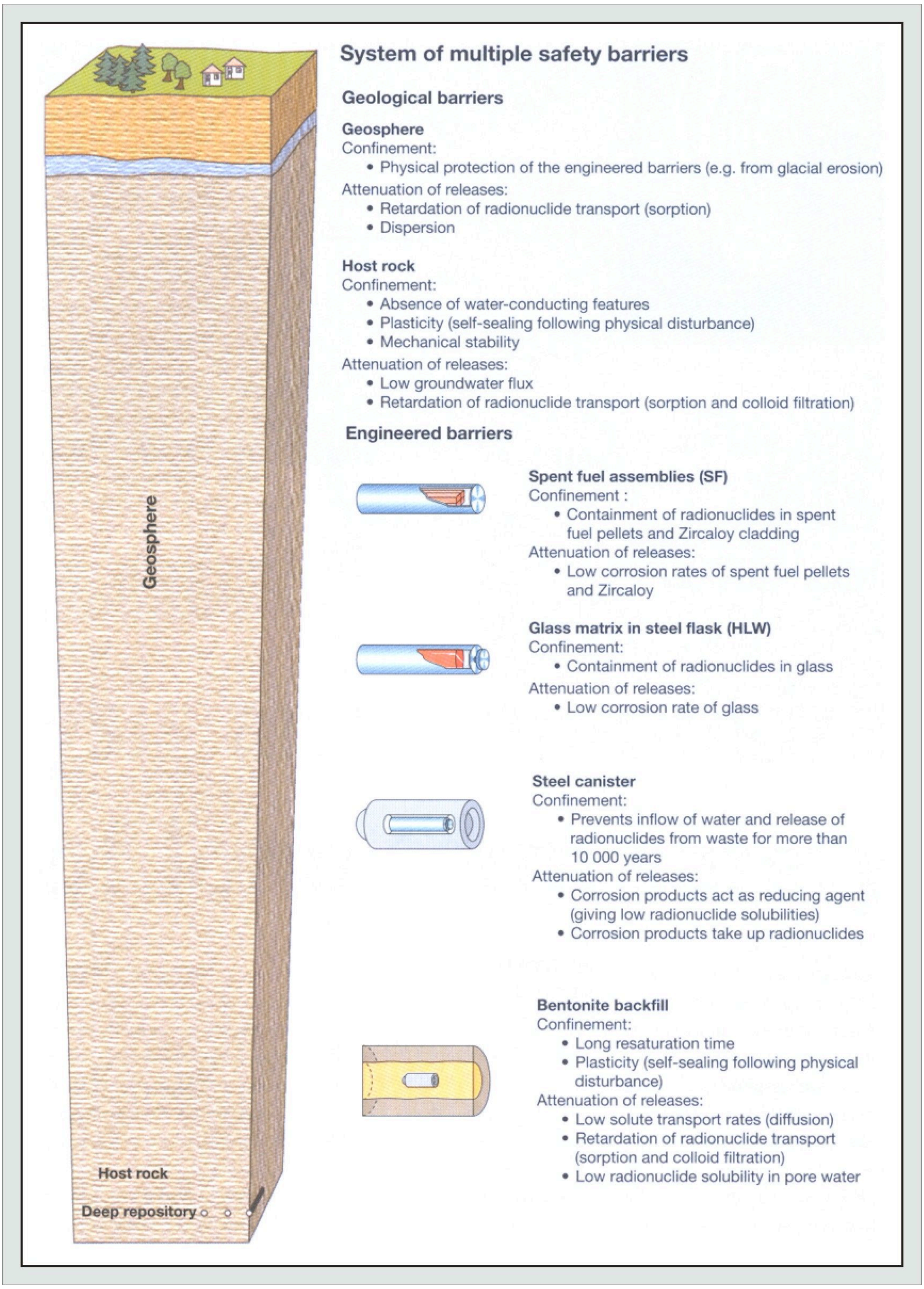


Fig. 5.14: A system of multiple barriers and their safety functions – in this case for the disposal of HLW and SF in Opalinus Clay in Switzerland

5.5.5 Worldwide experience and RD&D

The designs most commonly considered for deep geological repositories envisage mined openings, at depths between about 200 m and 1200 m below ground, with emplacement of radioactive waste either in underground drifts or caverns, or in boreholes drilled from drifts. Usually a repository would be purpose-built but, in some countries, the use of existing mines has been proposed, and existing mines have been used for disposal of some categories of radioactive waste. The characteristics of the repositories are strongly influenced by the host geology, which affects both the mining techniques that are practical and the repository design, especially the dimensions of stable underground openings that are possible. In most countries, geological formations can be found that have the necessary characteristics (e.g. adequate depth, restricted-to-negligible groundwater movement, favourable geochemistry, suitable geotechnical properties) and have remained stable over millions of years. Most research has focussed on the development of repositories in the following rock types:

- plutonic or “crystalline” rocks (e.g. granite, gabbro and gneissic rocks),
- extrusive volcanic rocks (e.g. basalt, tuff),
- argillaceous sediments (e.g. clay, mudstone, marl, shale),
- and evaporites (e.g. halite (rock salt), anhydrite).

A summary of the positive and negative properties (advantages and disadvantages) of each of the above as potential host rocks for the development of deep geological repositories is given in [82]. Disposal in unsaturated rocks is under investigation at Yucca Mountain in the Nevada desert in the United States, but no similar environments exist in Europe where disposal would be in fully-saturated rocks. In each country, the decision to focus on a particular geological environment or type of rock is made on the basis of the available geological environments and other factors. Table 5.4 indicates deep geological repository options that have been, or are being investigated, in countries with well-developed geological repository programmes.

An up-to-date survey and analysis of the status of geological disposal options for HLW and SF that have been considered or are under investigation in major programmes – including Sweden, France, Germany, Belgium and Switzerland – is presented in [68]. A summary of the scientific and technical basis of geological disposal prepared by experts on behalf of the IAEA is given in [83], and a summary of RD&D within the EC 5th Framework Programme is given in [84]. An overall picture of elements of a geological repository, and summary descriptions of the wide variety RD&D projects in support of technical development and safety assessment of geological repositories, is given in [33].

Host rock types	Under active consideration in: *Also considered in the past in:*	Repository and EBS concepts
Fully saturated strong fractured rocks (e.g. granite, gneiss, basalt and tuffs)	Canada, Czech Republic, Finland, France, Germany (back-up option), Japan, Sweden, Switzerland (back-up option). *Spain, UK, USA.*	Emplacement of HLW and SF within clay buffer in vertical and horizontal boreholes, and drifts (horizontal tunnels). Emplacement of ILW / TRU within large caverns and silos, backfilled with cement materials.
Argillaceous sediments (e.g. clay, mudstone, shale)	Belgium, France, Japan, Switzerland. *Canada, Spain, UK, USA.*	Emplacement of HLW and SF with or without buffer in vertical and horizontal boreholes, and drifts. A clay buffer is most common but a cement buffer is considered in the current Belgian concept. Emplacement of ILW / TRU within tunnels, backfilled with cement materials.
Salt or anhydrite	Germany, USA. *France, Netherlands, Spain, UK.*	Emplacement of HLW and SF in narrow vertical and horizontal boreholes without backfill and emplacement in drifts with crushed salt backfill. Emplacement of ILW / TRU within large caverns and silos in salt, backfilled with crushed salt (MgO is used at the WIPP facility), and within tunnels in anhydrite.
Unsaturated volcanic tuff	USA.	Emplacement of HLW and SF in large diameter drifts, without backfill.

Table 5.4: Indication of deep geological repository options that are being, or have been, investigated worldwide.

The above referenced reports [68, 83, 84, 33], and the reports of the individual national programmes, illustrates that:

- A detailed scientific understanding has been developed of the processes most relevant to the design, performance and long-term safety of geological repositories. This is based on over 30 years work in materials science, chemistry, earth sciences and many more specialist disciplines.

- The general requirements to ensure the long-term safety of a geological repository for radioactive waste are understood, and it has been shown that these can be provided in a range of geological settings.

- Properties of potential repository sites, and of rocks similar to those of potential sites, have been studied through extensive surface-based characterisation programmes and investigations in underground facilities. Thus, the techniques needed for site characterisation have been practised and their advantages and limitations examined, and an understanding of the processes operating deep underground has been developed and demonstrated, e.g. through modelling of the observed processes.

- A range of possibilities exists for design and construction of engineered barriers to complement the geological barriers and appropriate to the different types of radioactive waste. A sound and practical understanding has been developed of the properties of the relevant materials with respect to their fabrication and performance in a repository environment.

- The construction and short-term performance of key components of geological repositories have been demonstrated at model and full scale, e.g. the fabrication of waste forms and containers, and the underground emplacement of full-size containers. Projects now ongoing are investigating elements required to achieve emplacement of radioactive waste at the industrial scale.

- The accumulated scientific knowledge is sufficient to understand the long-term containment and migration of radionuclides in the engineered barriers and geosphere. Although uncertainties will always be present, these can be bounded sufficiently to be able to estimate the long-term performance of a repository system through quantitative models. Confidence in the results from these performance calculations is supported by evidence from natural analogues and long-term experiments in underground research facilities.

Almost all of the above knowledge and experience has been gained in open scientific and technical programmes, and much of it has come from multi-national collaborative programmes. This openness and collaboration allows for efficient dissemination of ideas and results, healthy discussion and critical peer review.

5.5.6 Implications for the safety of a shared European repository

Overall, through the extensive programmes of RD&D in several countries, the main technical issues related to the development of a deep geological repository and evaluation of its safety have been identified. Where there are outstanding uncertainties or developments needed these are being worked on, and the trend of RD&D and its results from various programmes worldwide give a high level of confidence that the development of deep geological repositories is technically feasible. Site-specific assessment studies within national programmes indicate that a well-sited geological repository can provide the required level of long-term safety.

In the USA, a deep geological repository for long-lived low and intermediate level waste has been licensed to receive waste and has been operating since 1999; a site has been selected and a licence application has been submitted for the construction of a deep geological repository for spent nuclear fuel and high-level waste. A deep geological repository for non-heat generating radioactive waste has also been licensed in Germany. The decision to proceed to detailed underground investigation of potential deep geological repository sites has been taken in several other countries, including Finland and Sweden in respect of repositories for spent nuclear fuel. The permission to take such steps towards siting and licensing has only been granted after thorough scientific and regulatory review, and with due regard to the cost and timescale of such projects. Several other countries, such as Belgium, France, Germany, Switzerland and Japan, are working towards being able to take such steps.

It can therefore be concluded that there is a high level of confidence in the feasibility and long-term safety of geological repositories for radioactive waste. Scientific and technical problems will still need to be solved, for example to understand some site-specific and waste-specific processes, and to implement some aspects of the engineering, but these problems do not detract from the basic soundness of the concept or its ultimate safety. Moreover, the combined effort being applied in several countries to develop and demonstrate actual facilities and their component parts provides excellent opportunities for collaborative, and hence cost-efficient, solutions to the scientific and technical challenges that would need to be met in developing a shared European repository.

5.5.7 Indicative levels of safety

Numerous assessments have been made of the post-closure performance and safety of geological repositories, including in recent years assessments based on site-specific data and detailed engineering concepts. In examining or comparing results from different assessments it must be borne in mind that:

(1) in different geological disposal concepts, safety may depend on different features and processes related to the different engineered barrier designs, host rock and environment;

(2) safety assessments are carried out with the information and understanding available at a given time to investigate safety and provide guidance with respect to forthcoming programme decisions or future work;

(3) different approaches may have been taken to treatment of uncertainties, and especially different degrees of caution (conservatism) incorporated into the analysis in the face of uncertainties that are either intrinsically irreducible or not reducible at the current level of data and understanding.

Thus, it is generally difficult to draw conclusions regarding the comparative performance of different options from independently developed assessments, see for example [85] and [86]. Comparison between alternative sites is only likely to be valid if the engineered concepts and assessments are as far as possible similar and take account of similar levels of data from each site, which is only likely to be the case with a well-developed national repository development programme, e.g. see [58].

Nevertheless some useful information can be gained from examination of post-closure safety assessments from advanced national projects, which is indicative of the level of safety that might reasonably be expected from an appropriately-sited and well-designed geological repository anywhere in Europe. Many assessments have been published; the following considers selected results from two assessments that illustrate some relatively generic features of such results.

Figure 5.15 shows results from the safety assessment of deep repositories for SF, HLW and ILW sited in Opalinus Clay of Northern Switzerland at about 650 m below ground [59]. These are the results of deterministic calculations of the dose as a function of time to a member of a hypothetical group living in the future for the Reference Case – which includes reference models and data.

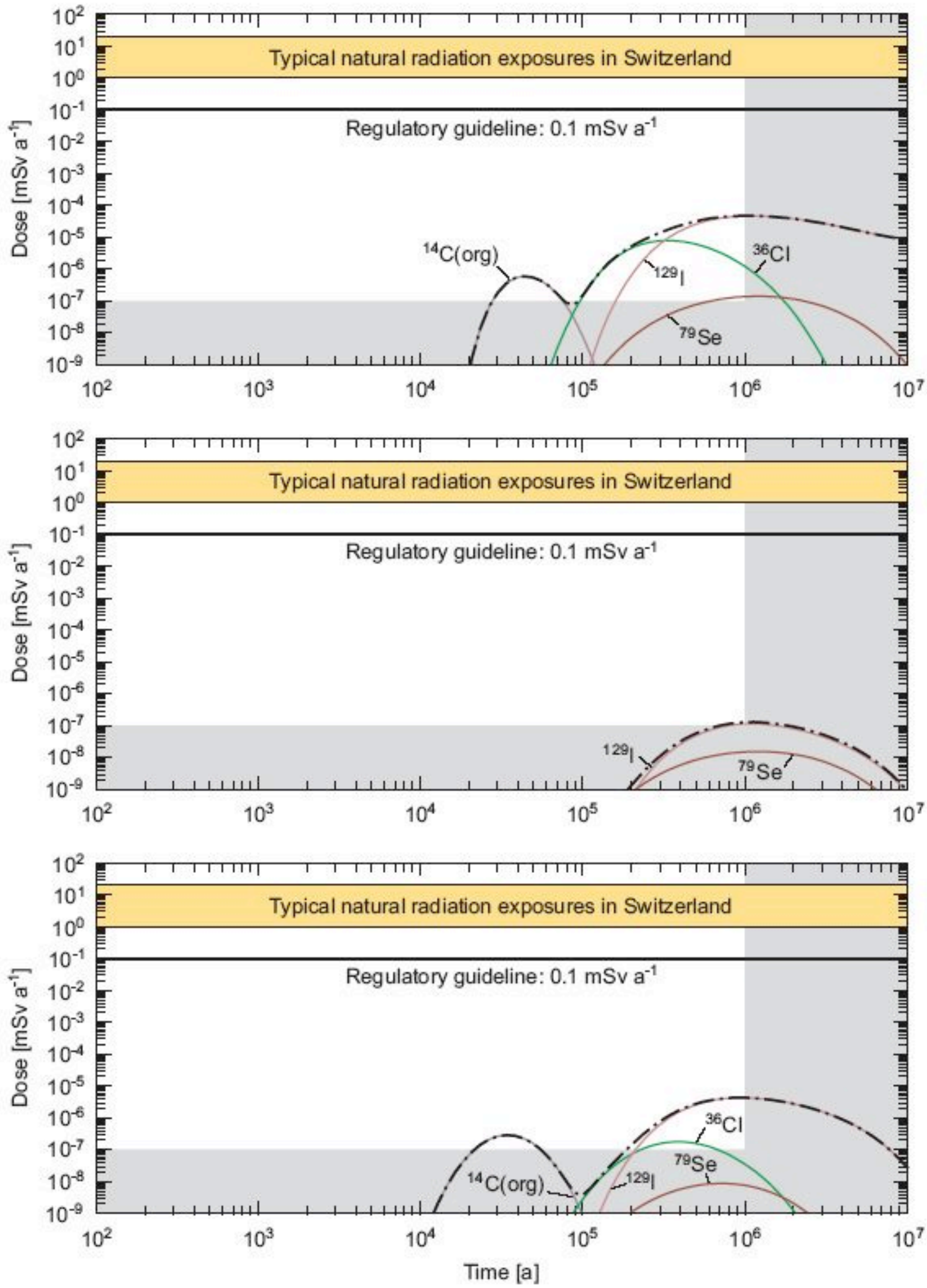


Fig. 5.15: Calculated doses for the Reference Case for disposal of SF (top), HLW (middle) and ILW (bottom) in Opalinus Clay in Switzerland [59]

Some features that can be observed are as follows:

- Assessed doses due to the repositories are zero for many years into the future.

In the case of SF (top figure) dose curves rise at about 20,000 years, that is 10,000 years after canister breaching which occurs at 10,000 years after closure. ILW curves rise somewhat earlier since the concrete waste packages do not offer such complete containment at early times.

- Assessed doses from SF are much higher than for HLW, with doses from ILW being intermediate.

This finding depends on the inventories considered, but also turns out to be a common finding from national assessments in which all types of waste are considered. The reasons are that (1) spent fuel contains the full spectrum of radionuclides, whereas HLW is depleted in more mobile radionuclides, e.g. C-14 and I-129, that are able to move with relatively little retardation in the engineered barriers or geosphere, (2) ILW contains the full spectrum of radionuclides but generally in lesser amounts, and (3) SF and HLW containers have longer lifetimes.

- Assessed doses are a small fraction of the regulatory guideline and even smaller fraction of the doses that would be received from natural radiation exposures.

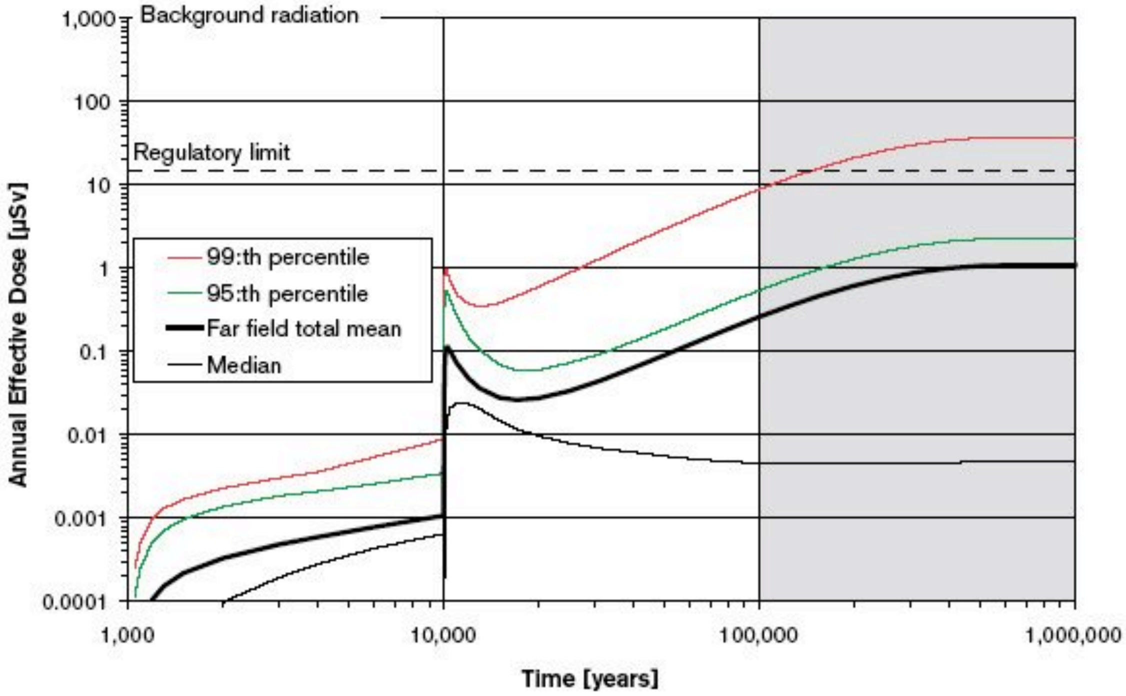
This tends to be a general finding because a deep repository project would not be proposed unless it could be shown with confidence that the assessed performance met regulatory guidelines, and regulatory guidelines are set at a low level well below the doses that would be received from natural radiation exposures.

- Maximum assessed doses are dominated by the more mobile radionuclides, i.e. those that are not solubility limited and not much retarded by sorption processes.

This depends on the disposal system. In some systems, uranium chain radionuclides may features at longer times, but in the case of results for disposal in the Opalinus Clay, shown here, are retained within the very low permeability host rock.

Figure 5.16 shows results from the safety assessment of deep repositories for SF sited in crystalline basement rock at two sites [12] in Sweden [58]. These are the results of probabilistic calculations of the annual dose (upper) and risk (lower) of as a function of time to a member of a hypothetical group. The upper figure illustrates a hypothetical case assuming the failure of a number of copper-shell canisters due to initial defects; this is a test case to examine the performance of the geosphere. The calculations take account of uncertainty in positions of the failed canisters and heterogeneity in flow pathways though the geosphere. The lower figure illustrates calculated risk taking account of canister failures due to shearing across deposition positions due to large earthquakes and canister failures due to corrosion if the buffer has been eroded by glacial melt waters. The calculations include several pessimistic assumptions.

[12] Forsmark and Laxemar, considering a repository located at about 400 and 500 m below ground, respectively.



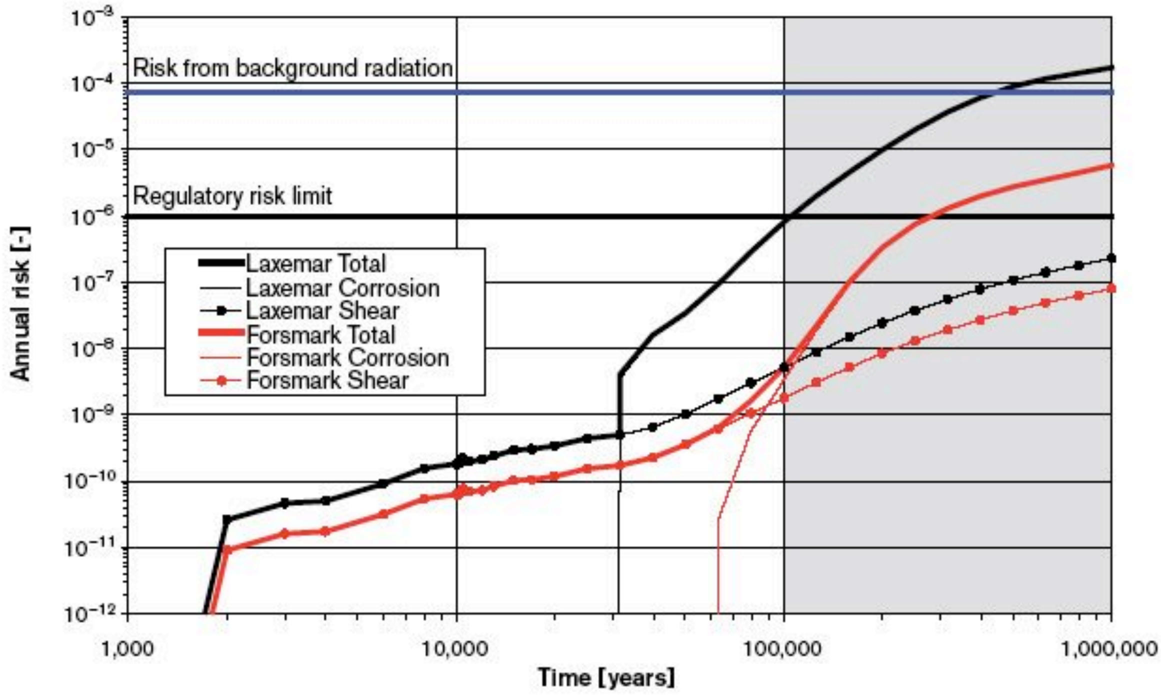


Fig. 5.16: Probabilistic calculations of annual doses for a test case assuming canister failures due to initial defects (upper) and for canister failures due to shearing and corrosion at two sites (lower) in Sweden [58]

Some features that can be observed are as follows:

- The range of results when uncertainties are taken into account (upper figure).

Most usually, the arithmetic mean of the dose distribution is taken as the best, unbiased single measure of the distribution; it is the sum of the probability of dose in each dose increment multiplied by the dose value. Further multiplying by the dose-risk factor yields the annual radiological risk (lower figure).

- Even with the pessimistic assumptions made, results in compliance with regulatory requirements can be achieved.

The SR-Can report remarks that the total calculated risk up to 100,000 years is at most close to regulatory limit at Laxemar and about two orders of magnitude below at Forsmark. The risk is pessimistically based on that calculated for the canister corrosion scenario, where several uncertainties are handled pessimistically. The risk calculated for Forsmark is based on a pessimistic interpretation of the current hydraulic situation. More recent site data from Laxemar indicate that the hydrogeological conditions are more favourable than those represented in the model used in SR-Can. It is, thus, concluded that the calculated risks for the two sites comply with the regulatory requirements during the initial glacial cycle after closure. Since SSI's general guidance indicates, "a strict quantitative comparison of calculated risk in relation to the criterion for individual risk in the regulations is not meaningful" for the period after the period beyond the initial glacial cycle, SKB conclude that calculated risks for the time beyond the initial glacial cycle also fulfil the regulatory requirements by comparison to risks from natural background radiation [58].

Results from post-closure safety assessment do not necessarily scale with inventory, but rather depend on characteristics of the engineered barriers and host rock, the spatial relationships of the repository, potential pathways though the geosphere and nature of the biosphere into which radionuclides may eventually emerge, and the degree of caution (conservatism) built into the assessment calculations. For example, in a comparison of results from five safety assessments all of which considered disposal of SF or HLW in crystalline rocks (see Figure 7.6 in [86]), all assessments calculated peak annual doses below about 0.01 mSv; the inventories considered varied between 1840 and 162,000 tU SF or tU equivalent for HLW and the peak dose rank order is quite different to the inventory rank order. This range encompasses the SAPIERR II small and large inventory situations of 6280 tU SF and 25,640 tU SF, see section 4.2.

The ICRP cautions specifically against attempting to calculate collective doses at long times into the future since the size of the exposed population becomes increasingly uncertain and the current judgements about the relationship between dose and detriment may not be valid for future populations [22]. Hence, collective dose is "is of limited use in the context of the disposal of long-lived radioactive waste … However, consideration of the number of people potentially involved and the distribution of individual dose in time can be of some help" [23].

5.6 Safety overview and comparative radiological impacts

5.6.1 Safety overview

The previous sections of this chapter confirm that radiological safety is achievable for all steps required within a European shared waste management system as discussed in Chapter 4.

This has been demonstrated in practice for the steps of radioactive waste handling, transport and storage, including for spent nuclear fuel. The step of sealing of SF/HLW into disposal containers has not been demonstrated, but appropriate technologies and have been tested in Sweden, Germany and elsewhere, and the radiological protection measures are the same as those already in use for handling SF/HLW. ILW is routinely packaged for storage and disposal in many countries. Radiological safety assessments and practical experience shows that the necessary steps can be safety accomplished in accord with international guidance and in compliance with national laws and regulations.

Licensing and operation of a deep geological repository for SF, HLW or long-lived ILW has not been demonstrated, but several countries are working towards that goal, and there is every reason to believe that such facilities will be brought into operation within the next two decades. Demonstration of the long-term safety of a geological repository will remain challenging, but probably less challenging than the political and social challenges associated with such developments [87].

Radiation standards and indicative doses for transport, facility operations and the post-closure period have been discussed in previous sections of this chapter. A key element is that a national or an international radioactive waste management system will be designed to meet design targets that will be set to ensure that individual and collective doses and risks are ALARA. This will be achieved by:

- application and assurance of good management and technical practices, including application of best practical means (BPM) and thorough quality assurance programmes;
- strategic application of constrained optimisation based on iterative safety and design studies, i.e. careful decision making with safety in mind at each step of the design and implementation process, including concept development, siting and operations;
- confirmation of optimisation or choice between specific technical options aided by qualitative and quantitative analyses, e.g. multi-attribute analysis (MAA), safety analyses and cost-benefit analyses (CBA);
- monitoring of radiological safety and technical performance;
- effective regulatory scrutiny and oversight within a strong legal and regulatory framework.

Hence, each national and any shared European waste management system will be made safe as judged against international standards and the relevant European Directives, and within the legal, political and practical constraints in each country.

5.6.2 Comparative radiological impacts – important caveats

A specific goal for this task within SAPIERR II is to indicate possible differences between radiological impacts that might be associated with development of shared European final disposal system for SF, HLW or long-lived ILW relative to the case in which each EU country would develop its own final disposal system.

Two very important caveats must be made that circumscribe any such comparison:

- Common international guidance and EU laws will apply to any national or international radioactive waste management system developed within the EU. High standards of safety will be demanded by society and national governments, sought by the developer and enforced by regulatory bodies. Hence any radioactive waste management system (national or international) developed in the EU will be safe – where this means as safe as it can reasonably be made (applying BPM and ensuring doses and risks are ALARA) and in compliance national laws and regulations.

- It is not possible to make detailed or firm estimates of the radiological impact of a system that exists only as a broad concept. Rather, in due course, radiological assessments will be needed based on more detailed descriptions of the actual facilities, possible locations, technical activities, transport routes, etc. Moreover, since the system will only be realised several decades from now, developments in technology and practice may change what is regarded as BPM, with the possibility that radiological risks and doses may be even further reduced.

With these caveats in mind, the following sections give indications of maximum individual radiation doses (section 5.6.3), indicative collective doses to workers (section 5.6.4) and indicative collective doses and dose commitments to members of the public (section 5.6.5), based on current nuclear industry practice and experience.

Indicative estimates of collective doses from a shared European waste management and repository system are compared to a base case in which each EU country would develop its own system. We focus on spent fuel (SF), because more radiological data is available for the steps needed for SF management, and considering the inventory (section 4.2), it is likely to be most important radiologically.

Radiological risks are not assessed. This is because to calculate risk requires estimation of the probability and radiological consequences of accidents or events that the system will be designed to avoid and hence will be very unlikely; see for example the discussion on transport accidents in section 5.3.5. Hence, the calculation of risk would be even more tenuous than calculation of dose, although some conclusions can be drawn about relative risks. Both national and shared waste management systems would be designed so that radiological risks due to accidents or very low probability events do not exceed the risks implied by dose design targets for normal operations.

5.6.3 Maximum individual doses

Safety will be assured as discussed in section 5.6.1, and since the required steps in a national or shared system are the same, we consider that a future shared European radioactive waste management system will be able to meet design targets similar to those that have been set for national systems based on application of BPM.

Table 5.5 presents illustrative design targets for a shared European waste management system that can also be taken as the maximum individual doses that will be incurred for normal operations and after closure.

These are based on the UK Nirex design targets for operations and transport (see Table 5.2), and typical targets for assessed dose for the post-closure period (see Table 5.1).

Discussion of indicative doses in previous sections confirms that these dose targets can be met. The illustrative design targets are compared to limits set in the Basic Safety Standards [20] and the EC Directive 96/29/EURATOM [17], and also to average levels of exposure to natural background based on data compiled by UNSCEAR, [88] Annex B. This indicates that maximum individual doses will be well below international limits and within the range of doses from natural sources.

<table>
<tr><th></th><th>Illustrative design targets</th><th>BSS/EURATOM limits [13]</th><th>Natural background [14]</th></tr>
<tr><td colspan="4">For facility operations and transport</td></tr>
<tr><td>Monitored radiation workers</td><td>2 mSv/a</td><td>20 mSv/a</td><td rowspan="3">Average
2.4 mSv/a

Range
1 to10 mSv/a</td></tr>
<tr><td>Other workers</td><td>0.5 mSv/a</td><td>5 mSv/a</td></tr>
<tr><td>Members of the public</td><td>0.2 mSv/a</td><td>1 mSv/a</td></tr>
<tr><td colspan="4">For the post-closure period of a repository</td></tr>
<tr><td>Up to 10,000 y
Calculated dose as a performance indicator</td><td>0.1 mSv/a</td><td></td><td rowspan="2">As above</td></tr>
<tr><td>From 10,000 to 1,000,000 years
Quantitative and qualitative arguments for continued safety</td><td>0.1 - 0.3 mSv/a</td><td></td></tr>
<tr><td>Beyond 1,000,000 years
Qualitative arguments</td><td></td><td></td><td></td></tr>
</table>

Table 5.5: Illustrative design targets for a shared European waste management system and comparison to accepted limits and natural background

5.6.4 Collective dose to workers

A simple model to calculate indicative collective doses for workers during spent fuel management is set out in Table 5.6. This is based on data already presented in sections 5.3.5, 5.4.6 and 5.5.7, and other data as indicated in the table.

[13] Limits set in the IAEA Basic Safety Standards and the EC Directive 96/29/EURATOM, which have been adopted into law as minimum standards in European countries.

[14] Typical levels of background due to natural sources, from Annex B, UNSCEAR 2000.

Waste management step (see Fig. 4.2)	Formula for collective dose	Basis
For wet storage	$CD_{wet} = 50 + 0.01 \times tSF_{cap}$ mSv/a	50/50 assumption for the facility, see text, 'calibrated' against dose of 100 mSv/a for CLAB [89].
For dry storage	$CD_{dry} = 5 + 0.001 \times tSF_{cap}$ mSv/a	50/50 assumption for the facility, see text, and 'calibrated' against doses of 10 mSv/a for CASCAD (dry vault) [89].
Cask loading and handling	$CD_{load} = 0.4 \times tSF_{tran}$ mSv	Loading CASTOR V19 (10 tSF) casks at Neckar [52], see section 5.3.5.
Transport	$CD_{trans} = 0.01$ mSv / 100 km / shipment	Transport of V19 casks from Neckar to Gorleben [52], plus estimate based on mean dose to drivers/crew of 0.01 mSv/h.
Cask inspection, unloading and SF encapsulation	$CD_{unload} = 0.4 \times tSF_{tran}$ mSv	As for cask loading. Cask unloading, transfer to and encapsulation in a number of disposal containers.
Repository operations and emplacement	$CD_{rep} = 5 + 0.001 \times tSF_{cap}$ mSv/a	20/80 assumption for the facility, see text, 'calibrated' against an estimate of 25 mSv/a for a 20,000 tU repository. (Mean dose of 0.25 mSv/a each to 100 staff, see section 5.4.6).

Note: tSF_{cap} = tonnes of spent fuel capacity; tSF_{tran} = tonnes of spent fuel transferred

Table 5.6: Model to calculate indicative collective doses to workers for the waste management steps for spent fuel (see Figure 4.2)

The model is based on limited data (see table) and simple assumptions. The key assumptions are as follows.

- 50% of the collective dose for a storage facility arises due general maintenance and operation of the facility and 50% is related to the capacity or number of transfers in or out per year, which is assumed to be in proportion to the capacity. This is arbitrary but not unreasonable. The calibration is against doses estimated at the CLAB assuming a capacity of 5000 tSF[15]. Radiological conditions at CLAB are very good compared to conditions at typical pool storage facilities at reactor sites where most spent fuel is currently stored (e.g. see data in [90]). Hence, the model may underestimate worker collective dose for typical wet storage facilities. On the other hand, the values for CLAB may provide a reasonable average for storage, assuming that most fuel remains in wet store but a fraction is transferred to dry vault or cask stores at which routine worker doses will be lower.

- Spent fuel is transported in CASTOR V-19 or similar flasks containing 10 tonnes of spent fuel each. Such large capacity casks are used, although a more typical cask capacity is

[15] We are aware that extension to 8000 tSF is planned.

3 tonnes [89]. The loading and unloading doses are insensitive to the assumption if collective dose scales with tSF transferred. Transport doses might increase for a larger number of smaller single cask shipments. On the other hand transport of several casks at a time is also possible and, if transport is by rail, crew doses may tend to scale per shipment rather than per cask.

- The doses from loading CASTOR casks described in [52], which involved fuel of unknown but possibly relatively young age, are representative of fuel loading or unloading operations of fuel at 30 to 50 years post irradiation as might be accepted for disposal. This is probably cautious, although it is possible that degradation in storage might cause changes that would in turn lead to higher doses during loading and unloading for some fuel.

- Only 20% of the collective dose for repository operation arises in relation to general maintenance and operation of the facility and 80% is related to the capacity or number of transfers in or out per year, which is assumed to be in proportion to the capacity. This assumes that SF disposal containers are emplaced and then backfilled progressively, so that the dose from disposed containers is zero and repository doses occur mainly in relation to emplacement operations. This is different to the 50/50 case of the storage facility where there is assumed to be some impact, e.g. on maintenance operations and monitoring, from the stored SF. Doses due to naturally-occurring radon daughter inhalation in the underground are not considered, although such doses are to expected and may be significant depending on the levels of uranium and thorium chain nuclides in the host rock.

Some further assumptions are needed to complete the calculation of collective doses. Table 5.7 shows the assumed data for notional small national systems and shared systems based on the scenarios discussed in Chapter 4; amounts of spent fuel have been rounded consistent with the limited accuracy of the dose model. Three national systems and two shared systems are specified, where capacities of several notional national systems can be combined to equate to the capacity of the small or large inventory cases discussed in section 4.2. The national system combinations used to form a shared system are somewhat arbitrary but are representative of actual possibilities as could be derived from in Table 1 in the SAPIERR I final report [2], which records the inventory of spent fuel stored in the ten SAPIERR I countries in 2040.

Storage in national facilities (wet store, dry vault or dry cask stores) is not part of the shared system. On the other hand, an important benefit of the shared system is that it will be economic to bring a shared repository into operation sooner, considering the sum of SF accumulated in several national programmes. Hence, the time spent in national storage facilities may be decreased and it may be possible to avoid increases in storage capacity that would otherwise be needed. We understand that the possibilities here are various and complex, and would depend on the schedule of spent fuel arising in the participating countries and the date at which repository operations could begin. Nevertheless, we think this benefit should be represented and have done this by making an arbitrary assumption that the time spent in national storage facilities is reduced by 10 years on average.

Transport distances are as discussed in section 4.4.

	National spent fuel management systems			Illustrative shared systems	
Capacity, t SF	1000 t SF	2500 t SF	4000 t SF	6000 t SF	25 000 t SF
Arising from	national	national	national	3 countries	10 countries
Notional make up for shared system				1 x 1000 2 x 2500	3 x 1000 4 x 2500 3 x 4000
Time in national storage	50 years	50 years	50 years	40 years	40 years
Transport distance	100 km	100 km	100 km	400 km	1200 km
Repository operating period	10 years	15 years	20 years	25 years	50 years

Table 5.7: Data for notional small national spent fuel disposal systems and illustrative European shared spent fuel disposal systems

The encapsulation step is a significant “bottleneck” of repository operations. In Sweden, SKB plan to encapsulate and emplace “one canister per day”, which allowing 240 working days and some allowance for maintenance or other stoppages implies about 200 to 220 disposal containers per year. Depending on fuel type, each could contain about 1.5 to 2.0 tSF indicating a maximum emplacement rate of about 300 to 400 tSF per year. For a larger repository, i.e. the large inventory shared case, 2 encapsulation lines might be built. In any case, it could generally be assumed that although a small national repository might be built later, it could be filled and closed in a shorter period of time. This is represented in the assumed repository operating periods.

Applying the model described in Table 5.6 to the data in Table 5.7 yields indicative estimates of collective dose to workers as given in Table 5.8. Given the simple model and assumptions, the collective dose estimates should not be taken as reliable estimates for future SF management options. We consider, however, that they are indicative of the order of magnitude of possible doses based on current practice, and are sufficient for comparative purposes.

In relation to occupational doses from nuclear power, the UNSCEAR 2000 Annex E [88] remarks: “in the dose data currently available, the data specifically associated with waste management are rarely identified separately.” and does not give estimates of occupational doses related to waste management. Annex E does, however, give data related to occupational doses from reactor operations. In particular, Table 8 of Annex E gives the collective effective dose per unit energy generated for different reactor types over 5 year periods since 1975. Taking the period 1985 to 1994 as representative, and assuming a mix of 75% PWR and 25% BWR, the UNSCEAR data indicates a collective effective dose to reactor workers of 4.5 man-Sv per GW.a. Assuming a mean energy yield of 35 GW.d / tU, the 6000 and 25,000 t considered in Table 5.7 equate to 575 and 2400 Gw.a, and thus to collective effective doses to reactor workers of 2,600 and 10,800 man-Sv.

	National spent fuel management systems		Shared spent fuel management systems	
See text for assumptions	3 countries	10 countries	Small inventory	Large inventory
	6 000 t SF	25 000 t SF	6 000 t SF	25 000 t SF
	Collective doses, Sv (rounded)			
Storage				
if wet storage	11	38	8.4	30
if dry storage	1.1	3.8	0.84	3.0
Cask loading	2.4	10	2.4	10
Transport	0.006	0.025	0.024	0.30
Unloading and encapsulation	2.4	10.0	2.4	10.0
Repository operation	0.29	1.2	0.28	1.5
Total from loading to disposal	5.1	21.2	5.1	21.8
Total including national wet storage	15.6	59	13.5	52
Net dose saving			2.1	6.9

Table 5.8: Comparison of indicative collective doses to workers for individual national and shared spent fuel management systems of the same total capacity

The first observation on Table 5.8 is that the doses indicated are very small compared to collective doses related to the corresponding reactor operations (2,600 and 10,800 man-Sv, see above). Beyond this the following points emerge.

- After storage, the largest doses are associated with transport cask loading/unloading, which are independent of whether spent fuel is managed on a national or shared basis.
- Doses due to repository operations are small and vary little between the shared and equivalent capacity national options. This is because doses scale mainly in relation to the number of containers disposed.
- Doses due to transport between facilities are tiny, so that the higher dose in the shared case makes no impact on the total.
- The net dose saving indicated (in the last row) arises entirely from the assumption that a shared repository might reasonably be brought into operation earlier than most national repositories, so that time in storage and doses associated with storage are reduced.

It can be concluded that the consideration of indicative collective doses to workers does not present a significant discriminating factor between individual national and shared spent fuel management systems of the same total capacity. The estimated net dose saving is only about 1/1000th of the collective doses related to the corresponding reactor operations and arises from the assumption that early development of a shared disposal facility would reduce the time that spent fuel is stored at national disposal facilities.

5.6.5 Collective doses to members of the public

Information on individual or collective doses to members of the public from spent fuel management is even more sparse than that related to doses to workers. This is because, at present:

- most spent fuel is located in pools at reactor sites, and discharges to the environment and doses to the public are generally assessed for the whole site, which is dominated by reactor operations;
- there is limited experience, and no assessments that we have located, of discharges to the environment and local doses from either away from reactor (AFR) stores or geological repositories for SF.

Information is available for SF reprocessing but this is not relevant here. Some information is available from the UK Nirex programme (which can be interpreted to some extent as to its relevance for SF management) and from the ANDRA programme. Nevertheless, an indication of collective dose to members of the public can be developed by, first, identifying those steps in SF management that have most potential for doses to members of the public and, second, estimating collective doses from those steps. Regarding the steps:

- Spent fuel storage buildings are sited and designed so that external radiation levels at the site boundary will be negligible, and thus the external doses to the public must be very low considering the nearest habitation will be some distance for the site boundary.

- There will be some environmental release and consequent public dose related to the storage of spent fuel in pools. This can be said based on the facts that radionuclides can be measured in storage pool water and buildings, and pool building air is discharged via filtered stacks, which will allow the discharge of noble gas radionuclides and radionuclide-labelled gases. Releases and public exposures related to dry vault stores are liable to be much lower, and those related to dry cask stores effectively zero.

- By comparison to the case of pool storage, any release and public dose related to cask loading, unloading, and encapsulation, must be negligible. The activities will be carried out within contained and shielded areas; given the short periods of time involved, doses outside the site boundary must be effectively zero.

- Small doses to the public will arise from external exposure along transport routes. Release of radionuclides from sealed transport casks is zero and surface contamination levels are controlled so that any release from this source will also be effectively zero.

- SF disposal containers are likewise sealed so that the radioactive release from a repository related to the SF [16] will be near zero. External doses at the site boundary due to the SF will also be zero, see section 5.4.6.

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16 Radon emanating from rocks will be discharged with mine ventilation exhaust air. The increment to radon in local air will be negligible compared to the natural background level, see section 5.4.6.

- Results from repository post-closure safety assessment, e.g. see section 5.5.7, indicate that for a well-chosen site and suitable engineered barrier systems, the release to the environment will be zero up to at least about 10,000 years. There may be some release at longer times, but the ICRP has specifically advised against calculating collective doses due to releases at such long times.

Thus, we conclude that only two exposure pathways can contribute significantly to collective dose to members of the public from SF management:

- exposures due to the discharge of radionuclides from pool storage facilities, and
- exposures due to external irradiation along the transport routes.

Collective doses from pool storage facilities

At some pool storage facilities, doses might arise from the discharge of liquid effluents associated with pool water cleaning. These are liable to small, however, as in principle any radioactive wastes from pool water cleanup and maintenance should arise as ion exchange resins which are disposed as solid LLW. Doses due to aqueous discharges are not assessed here, because the doses are liable to be smaller than those due to airborne discharges and the doses will be sensitive to how the discharges are managed, e.g. into what size of water body, which is site-specific.

A calculation has been made of collective doses and collective dose commitments due to the airborne discharge of radionuclides from spent fuel pool storage facilities. The full details of the evidence and data used are complex and not presented here. The basic line of evidence and argument is as follows.

UNSCEAR 2000 [88] Annex C gives estimated collective doses from reactor operations according to models outlined in Annex A. UNSCEAR calculates collective doses in local and regional compartments (up to 2000 km from the site), which are due to the initial dispersion of radionuclides, and collective dose commitments up to 10,000 years from radionuclides that become distributed globally. We have used UNSCEAR data to calculate normalised collective doses and collective dose commitments assuming power production 75% by PWR and 25% by BWR reactor types. The total collective doses and their sources thus calculated are given in Table 5.9.

Collective dose	man-Sv / GW.a	Breakdown
in local and regional compartments	0.21	43% particulates[17], 38% ^{14}C, 18% noble gases, <1% ^{3}H
commitment due to globally dispersed radionuclides to 1000 years [18]	3.34	99.9% ^{14}C
commitment due to globally dispersed radionuclides to 10,000 years	20.5	100% ^{14}C

Table 5.9: Collective dose and collective dose commitments from airborne releases from LWR reactors per unit electrical energy

[17] The particulate emissions come mainly from BWRs.

[18] By adjustment of UNSCEAR commitment to 10,000 years, taking account of half-lives of the globally dispersed nuclides (^{3}H, ^{14}C, ^{85}Kr, and ^{129}I).

Considering that in SF pool building conditions particulates will not be discharged to air and the short-lived noble gases (primarily ^{133}Xe) will be absent, it can be concluded that collective doses due to the airborne discharges from SF pool storage will be almost entirely due to ^{14}C and only this radionuclide needs to be considered further.

A thorough review of the life cycle and management of carbon-14 from nuclear power generation is given by Yim and Caron [91]. In LWR types, ^{14}C is produced by neutron irradiation of nitrogen impurities in the UO_2 fuel, zircaloy cladding and other metal parts ($^{14}N(n.p)^{14}C$ reaction), and by irradiation of oxygen in the water coolant and UO_2 ($^{17}O(n.\alpha)^{14}C$ reaction). Yim and Caron [91] derive the normalised production rates of ^{14}C, which can also be compared with normalised reactor discharges from UNSCEAR [88] as shown in Table 5.10.

	PWR	BWR
	TBq ^{14}C / GW.a	
Production (from [91]):		
Fuel, UO_2	0.71	0.73
Zircaloy and fuel assembly parts	0.38	0.51
Coolant, H_2O	0.26 to 0.41	0.65 to 0.97
Total	1.3 to 1.5	1.7 to 2.0
Release (from [88], 1990-1994)	0.22	0.51

Table 5.10: Normalised production rates and releases of carbon-14 in LWR types

This confirms that the majority of ^{14}C produced in the water coolant is released during reactor operations; the remaining fraction goes into ion exchange resins and is disposed of as low-level waste. The activity produced in the fuel, zircaloy and fuel assembly parts will be retained with the spent fuel assemblies, which are stored. Table 5.10 indicates that the normalised inventory held in spent fuel storage is about 5 times that discharged from reactor operations, assuming 75% PWR and 25% BWR.

Both Yim and Caron [91] and Van Konyenburg [92] consider that some of this inventory will be become dissolved in the water of spent fuel pools, followed by exchange with ventilation air and exhaust from the stack to the atmosphere. This is based on three independent observations in the literature cited by Van Konyenburg, including direct measurement of $^{14}CO_2$ above a spent fuel pool. In addition, an example of pool water analysis reproduced by IAEA in [90] indicates ^{14}C as present at 30 Bq/litre, approximately 1/100th the level of dominant nuclides [19]. Van Konyenburg suggests the mechanism for release is corrosion of the zircaloy cladding, but considering the very low corrosion of zircaloy in pool conditions [90] this mechanism appears to be quantitatively trivial provided pool chemistry is well controlled.

Other possible sources of ^{14}C release could be dissolution of activated crud deposits formed during time in the reactor, and releases from leaking fuel elements. The rate of fuel element failures is very low and strenuous efforts are made to identify leaking fuel elements during reactor service and in storage (by ultrasonic, eddy current and sipping tests, e.g. see [93]);

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19 The IAEA "example" analysis shows ^{3}H 0.45, ^{14}C 0.034, ^{60}Co 5.64, ^{63}Ni 2.19 and ^{137}Cs 3.94 Bq/ml (selected nuclides).

still some release from this source is inevitable, e.g. see [90]. An additional issue that may need to be considered is oxidative splitting of fuel rods [94], which has been hypothesised as a mechanism for gross failure of cladding in fuel with existing pinhole leaks or hairline cracks that are exposed to an oxidising atmosphere, e.g. during transfer operations or in dry storage.

For the purpose of our calculations, we assume a total release during 50 years of storage of 0.1% (or 1/1000th) of the total ^{14}C inventory contained in the fuel assemblies. This is about 10 to 100 times the amount that could be accounted for by uniform corrosion of zircaloy in well-controlled pool conditions, but allows that conditions may sometimes deviate and that ^{14}C is most probably also coming from dissolution of crud and release from a small fraction of leaking fuel rods.

Allowing that the inventory of ^{14}C in the fuel assemblies is 5 times the inventory that is released during reactor operations (see Table 5.10), then the normalised collective doses from the release during storage will be 5/1000th of the normalised collective doses attributable to ^{14}C from reactor releases (see Table 5.9). We take the normalised collective doses due to 40 years of storage as 4/1000th of those from the reactor releases.

Collective doses from transport

To calculate the collective dose to members of the public due to external exposure from SF cask shipments, we consider the exposure in a strip on either side of the transport route.

Dose rates from individual transport casks will be limited to 0.1 mSv/h at 2 metres distance. Tunaboylu et al. [53] calculate a dose of 0.025 microSv to a person standing at 10m from the route of a container passing at 20 km/h, representative of road transport. We scale this to 0.010 microSv along the route of a container passing at 50 km/h, which we consider representative of rail transport.

This dose is taken as representative of the average dose to persons in a strip of 50m on either side of the transport route (i.e. 100m wide), with doses outside that strip being negligible. Assuming 200 persons per square kilometre, which is average for central Europe, this results in a collective dose of 0.010 mSv per 100 kilometre per shipment by rail or 0.025 mSv per 100 kilometre per shipment by road. We consider that transport should be arranged primarily by rail both to reduce the probability of accidents (see section 5.3.5) and for security reasons (see Chapter 6) and hence select 0.010 mSv per 100 kilometre per shipment. Assuming shipments by 10 ton cask this is, coincidentally, the same as the estimated collective dose to the transport crew, see Table 5.6. This seems consistent given that we are considering a larger number of persons per kilometre (20 per km instead of 2) but at somewhat greater average distance.

Indicative collective dose comparison

The indicative collective dose comparison is shown in Table 5.11. This shows zero doses for the stages of cask loading, unloading, repository operations and postclosure as argued above, and calculated doses due to storage and transport.

Assuming a mean energy yield of 35 GW.d / tU, 6000 and 25,000 tSF equate to 575 and 2400 Gw.a. Applying data in Table 5.9 derived from UNSCEAR, this indicates collective doses due to the associated reactor operations of 120 and 500 man-Sv in local and regional compartments and collective dose commitments to 1000 years of 2,000 and 8,000 man-Sv due to globally dispersed radionuclides.

	National spent fuel management systems		Shared spent fuel management systems	
See text for assumptions	3 countries	10 countries	Small inventory	Large inventory
	6 000 t SF	25 000 t SF	6 000 t SF	25 000 t SF
	Collective doses, man-Sv (rounded)			
Storage (from ^{14}C) Local and regional Global to 1000 years	 0.23 9.6	 0.95 40	 0.18 7.7	 0.76 32
Cask loading	0	0	0	0
Transport (local)	0.006	0.025	0.024	0.30
Unloading etc.	0	0	0	0
Repository operation	0	0	0	0
Post-closure to 10,000 years	0	0	0	0
Total incl storage Local and regional Global to 1000 years	 0.23 9.6	 0.97 40	 0.21 7.7	 1.06 32
Net saving Local and regional			0.02	- 0.09
Net saving Global to 1000 years			1.9	8.0

Table 5.11: Comparison of indicative collective doses to members of the public for individual national and shared spent fuel management systems of the same total capacity

The first observation on Table 5.11 is that the doses indicated are small compared to the collective doses to members of the public related to the corresponding reactor operations (120 and 500 man-Sv local and regional and 1000 year global commitments of 2,000 and 8,000 man-Sv). Beyond this the following points emerge.

- Total local and regional collective doses, and global dose commitments to 1000 years, are dominated by doses related to storage.

- Collective doses due to transport between facilities are relatively smaller, so that the higher dose in the shared cases make minor impacts on the local and regional total.

- The net dose differences indicated for local and regional collective dose (penultimate row) are very small. An apparent dose increase is calculated for the large inventory case, increased transport doses outweighing reduced storage doses, but the difference is probably less than the associated uncertainties, especially related to unknown conditions and duration of future storage.

- Calculated dose commitment from globally dispersed radionuclides, and therefore dose differences, are entirely related to discharges from storage facilities.

It can be concluded that the consideration of indicative collective doses to members of the public does not present a significant discriminating factor between individual national and shared spent fuel management systems of the same total capacity. The estimated net saving in global collective dose commitment to 1000 years is only about 1/1000th of the collective dose commitment related to the corresponding reactor operations, and arises from the assumption that early development of a shared disposal facility would reduce the time that spent fuel is stored at national disposal facilities.

Additional perspective can be gained noting that carbon-14 is produced naturally by the stratospheric irradiation of nitrogen from the $^{14}N(n.p)^{14}C$ reaction. ^{14}C atoms are readily converted to $^{14}CO_2$, which distributes in the atmosphere and is available for incorporation into the food chain by photosynthesis. The total inventory of ^{14}C due to natural sources has been estimated at about 200 PBq in the atmosphere and 10,000 PBq in the terrestrial environment [91]. Up to 1990, about 220 PBq was injected into the atmosphere from nuclear weapons testing, but will now be distributed in terrestrial and atmospheric reservoirs. UNSCEAR 2000 Annex C gives the worldwide time-integrated release of ^{14}C from reactors and reprocessing plants up to 1997 as 2.8 PBq, with release at a rate of 0.09 PBq/a in the period 1995-1997 [88]. Thus the released ^{14}C inventory, and hence collective dose impact, from all nuclear cycle operations worldwide is only a small fraction of that from natural sources. Indeed, the global dose impact is much less than the diluting effect of burning fossil fuel, which injects stable carbon (98.9% ^{12}C and 1.1% ^{13}C) into the atmosphere.

5.7 Final remarks and conclusions on safety

5.7.1 Balance of radiological impacts

The preceding sections show there is little difference between calculated radiological impacts for a large or small inventory shared European spent fuel management system and several national systems with equivalent capacity. The most important quantitative difference, or potential dose reduction, arises from the assumption that timely development of a shared repository would reduce the average time that spent fuel is stored at national facilities, especially wet storage facilities. Even so, the calculated collective dose reductions (to workers and to members of the public) are only about 1/1000th of the collective doses from the reactor operations that produced the spent fuel.

In these dose comparisons, post-closure radiological impacts do not figure because, for an appropriately sited and well-designed geological repository, no releases to the environment are expected until many thousands of years after closure, see section 5.5.7. We believe this is the correct perspective on foreseeable radiological impacts from spent fuel management, and correctly assesses the relative radiological impacts of shared versus national disposal.

5.7.2 Long-term safety as a special factor

On the other hand, the specific aim of geological disposal is to provide assurance of safety over very long times, up to the limit of geological stability which may be the order of several millions of years. Thus, even if a tangible radiological benefit cannot be shown, it is worthwhile to consider whether a shared system offers any advantage in this respect.

Arguments favouring a shared repository	Comment or counter argument
(1) Siting across a larger geographical area, several countries, will give a larger choice of potential host rocks and suitable sites. Hence, a better geological site can be selected.	Suitable geologies and sites can probably be found in all countries European countries with nuclear waste. The wider siting opportunities should give more technically viable options, but other issues will impact on siting and may be more important determinants than geological quality.
(2) Siting across a larger geographical area, several countries, will allow possibilities for siting further away from centres of populations or in regions of lower population density. Hence, doses from eventual releases from the repository will be lower.	In the time scale of a several hundreds to thousand years relative population densities do show some stability, but this is not so over thousands of years, for example as climate changes. In the long-term, the population characteristics of any locality cannot be estimated, and assessments are carried out on the basis that humans will be present, making use of local resources.
(3) There would be less chance that location shared European repository would be “forgotten” relative to one of several national repositories. Hence, the probability of inadvertent intrusion is reduced.	The locations of a shared and of national repositories will be recorded in national, European and international documents and archives (e.g. maps and information related to safeguards). Any disruption or change in perspectives sufficient to cause the loss of such knowledge would affect shared and national repositories.
(4) If knowledge is lost, there would be less chance of intruding on a single shared repository than one of several national repositories because: (a) it presents a smaller (single) target and (b) it can be sited in location of lower geological resource interest.	The area covered by a repository is related to geotechnical factors for all excavations, and to thermal considerations in the case of SF/HLW, so that a shared repository will cover an area similar to the summed area of equivalent capacity national ones. Thus, if inadvertent intrusion is viewed as a random event there is no difference. It may be possible to site a repository in an area perceived as lower resource interest from today’s perspective, but we cannot foresee future perspectives. See also argument (1) and response.
(5) A shared repository would be designed and implemented within a co-operative framework in which financial costs and technical expertise are shared. This provides a stronger financial and technical basis for high quality implementation.	Both national and shared repositories will be developed and implemented under protected fiscal plans. The line of responsibility is clearer in a solely national programme. National programmes can contract aspects in which they do not have expertise to foreign experts.
(6) A shared repository would be implemented under greater international scrutiny with assured regulatory oversight through the multinational agreements. Hence, a better quality of implementation is assured.	Both national and shared repositories will be developed and implemented consistent with common international standards. An increased multinational oversight and peer review could improve implementation, but the benefits may be related to confidence and transparency rather than actual long-term performance.

Table 5.12: Arguments concerning the relative long-term safety of a shared geological repository vs several smaller national repositories

Table 5.12, left hand column, sets down some of the arguments that have been suggested as to why a shared geological repository might offer long-term safety advantages over several smaller national repositories; counter arguments and remarks are given in the right hand column. Variants on the arguments can be thought of, but the six suggested capture the general line of possible arguments and counter arguments.

Considering the arguments and counter arguments in Table 5.12, the strongest possible arguments seem to be related to:

- greater choice of geological situations and sites – argument (1);
- a larger pool of financial and human resources – argument (5);
- greater international and/or multinational scrutiny – argument (6).

These are explored further in the following sections.

5.7.3 Choice of geological situations and between sites

The greater choice of geological situations and sites available over several countries could provide better opportunities for finding a geologically "better" site than might be found in some smaller countries. On the other hand, the international boundary conditions and safety standards will be the same for national and shared repositories and similar high standards of performance and demonstration of long-term safety will be required in both cases, see Chapter 3 and section 5.1.

It should be emphasised that it will not be the intention to find a "best technical site". This is partly because this is not possible without making extensive investigations of multiple sites, but mainly because it is not needed. Scientific investigations and assessments have shown that the required stringent levels of long-term safety can be achieved in a range of geological situations and sites, see section 5.5 and supporting references.

It is important that candidate sites meet certain technical criteria, for example regarding geological stability, host rock quality and volume, low groundwater flows, suitable geochemistry, potential for investigation and development. Given several sites that are estimated to provide suitable conditions, the choice between sites is likely to rest on other factors such as:

- suitable location with respect to transport infrastructure;
- consistency of the development with local and regional planning;
- availability of necessary technical expertise and work force, e.g. nuclear and/or mining expertise;
- developed regulatory system and capability;
- social and political factors.

Ultimately, local and regional public acceptability may be key.

We thus conclude that a shared repository project could give better options for technical site selection, but this advantage may not be fully realised. Technical site quality will be an important factor, but may not be the determining factor.

5.7.4 International and multinational scrutiny and collaboration

A shared repository would be designed and implemented under international and multinational scrutiny. On the other hand, national programmes also have access to international and multinational expertise and advice through the EC, NEA and IAEA and bi-lateral agreements. Ultimately, a national or a shared repository will be implemented under the legal standards and regulatory oversight of the hosting country. Any national regulator will, rightly, act independently within their national framework and only issue licences against a case that they judge satisfactory. Hence, it is difficult to assert that a tangible improvement in safety will be assured. As noted in Table 5.12, the benefits may be related to confidence and transparency rather than actual long-term performance.

Nevertheless, if the co-operative effort that can be mustered from several countries is well focussed, then it seems reasonable to suppose a fuller consideration of safety and technical issues can be made at each step, and a better quality of implementation achieved. It is important to note that this is a general argument in favour of a shared repository project that applies to all aspects of technical implementation, safety and security, not just long-term safety.

A further factor that can be added here, is that the combined efforts of several countries may give better prospect for joint realisation of a project at an earlier time than if national projects proceed independently. As indicated in section 5.6, this may provide a tangible safety benefit due to a reduction in the average time that spent fuel is stored at national facilities, and also a less quantifiable safety benefit of less chance that disposal will be indefinitely delayed in any country [20]. As remarked in the IAEA TECDOC on developing multinational radioactive waste repositories [95]:

> *"It is important to note that the improvements in safety and security that are expected are at a global scale. It is not intended to imply that a multinational repository will be safer or more secure than a properly implemented national repository. The global benefit results from making a proper disposal facility accessible also to countries that may not be in a position to implement a state of the art national repository."*

[20] A counter argument has been expressed in countries that are proceeding with geological disposal, that the prospect of a shared repository (that may or may not be realised) is itself a disincentive for countries to proceed with national projects.

6 Security

This chapter discusses physical protection aspects of security of a radioactive waste management system from waste acceptance to disposal. This includes discussion of nuclear security standards, defining and countering security threats, and physical protection systems in general terms. Physical protection of a shared waste management system and its stages are then discussed, and conclusions are drawn on the security of a shared system compared to a case of several smaller national systems.

This report does not address non-proliferation or nuclear safeguards in detail, since the controls are equally applicable to shared or national nuclear activities and under the same internationally-supervised arrangements, see section 3.4. These aspects are mentioned in section 6.4.2.

6.1 Security standards

As introduced in sections 3.1.3 and 3.3, the Convention on the Physical Protection of Nuclear Material [13], promotes international co-operation and common security standards for nuclear facilities and materials. Amongst other things, it obligates party States, which includes all members of the EU, to make specific arrangements and meet defined standards of physical protection for nuclear facilities and transport of nuclear material. These standards would be enforced on a shared European waste management system, and on national systems, through national obligations under the Convention and under the supervision of the responsible national regulatory and security offices.

The principles and implementation requirements are set out in the IAEA document on the Physical Protection of Nuclear Material and Nuclear Facilities (PPNMNF) [30]. This document sets out the objectives of the State's physical protection system:

- to establish conditions which would minimise the possibilities for unauthorised removal of nuclear material and/or for sabotage;
- and to provide information and technical assistance in support of rapid and comprehensive measures by the State to locate and recover missing nuclear material and to cooperate with safety authorities in minimising the radiological consequences of sabotage.

The PPNMNF describes elements of a State's system of physical protection of nuclear material and nuclear facilities, and sets general requirements related to:

- definition of a Design Basis Threat (DBT) developed from an evaluation by the State of the threat of unauthorised removal of nuclear material and of sabotage of nuclear material and nuclear facilities;
- legislation and regulation, including designation of competent authority, legal powers, licensing requirements, access to information and facilities, evaluation of threat;
- confidentiality, including protection of specific or detailed information the disclosure of which could compromise the physical protection of nuclear materials and nuclear facilities, and sanctions against persons violating confidentiality;
- evaluation of the implementation of physical protection measures to ensure measures are maintained in a condition capable of meeting the State's regulations and of effectively responding to the DBT.

The categorisation of nuclear material is then described, which is based on the quality and amounts of fissile material that could give potential for the construction of a nuclear explosive device by a technically competent group, see Table 6.1. Commercial spent fuel, typically enriched to a few % of ^{235}U, is category II, since although it contains a substantial inventory of radionuclides, the concentration of fissile elements is relatively low and specialised facilities would be needed to extract these for use in a nuclear weapon.

		Category I	Category II	Category III
Plutonium	Unirradiated	2 kg or more	Less than 2kg but more than 500g	500 g or less but more than 15g
Uranium-235	Unirradiated			
	- uranium enriched to 20% ^{235}U or more	5 kg or more	Less than 5 kg but more than 1kg	1 kg or less but more than 15g
	- uranium enriched to 10% ^{235}U but less than 20% ^{235}U		10 kg or more	Less than 10kg but more than 1kg
	- uranium enriched above natural, but less than 10% ^{235}U			10 kg or more
Uranium-233	Unirradiated	2 kg or more	Less than 2kg but more than 500g	500 g or less but more than 15g
Irradiated fuel [21]			Depleted or natural uranium, thorium or low-enriched fuel (less than 10% fissile content)	

Table 6.1: Categories of nuclear material

The PPNMNF describes requirements for *physical protection against unauthorized removal of nuclear material in use and storage*, which are graded according to the category of nuclear material. This includes consideration of physical protection through facility design, definition of protected and inner areas, limiting and controlling access, monitoring and patrols, predetermination of trustworthiness of all individuals permitted unescorted access to nuclear material or facilities, and response to intrusion threats.

The PPNMNF describes requirements for *physical protection against sabotage of nuclear facilities and nuclear material during use and storage*. These set down general requirements similar to those related to protection against unauthorized removal of nuclear material, and include specific requirements for nuclear power reactors. For other nuclear facilities, States should determine the level of protection needed against sabotage depending upon the degree of radiological consequences. In principle, encapsulation facilities and repositories offer much lower potential for radiological consequences than nuclear reactors or spent fuel wet storage facilities, but considering the sensitivity of such developments the host State might decide that a shared encapsulation plant and repository might be protected at the same level as a nuclear reactor site.

[21] The categorisation of irradiated fuel in the table is based on international transport considerations. The State may assign a different category for domestic use, storage, and transport taking all relevant factors into account.

Finally, the PPNMNF describes requirements for *physical protection of nuclear material during transport*, which again are graded according to the category of nuclear material. The document comments that transport of nuclear material is probably the operation most vulnerable to an attempted act of unauthorised removal of nuclear material or sabotage. Therefore, the physical protection provided should be “in depth”. Achievement of the objectives of physical protection should be assisted by:

- minimising the total time during which the nuclear material remains in transport;
- minimising the number and duration of nuclear material transfers, i.e. transfer from one conveyance to another, transfer to and from temporary storage and temporary storage while awaiting the arrival of a vehicle, etc.;
- protecting nuclear material during transport and in temporary storage in a manner consistent with the category of that material;
- avoiding the use of regular movement schedules;
- predetermination of the trustworthiness of all individuals involved during transport of nuclear material; and
- limiting advance knowledge of transport information to the minimum number of persons necessary.

Guidance on the implementation of the requirements is given by the IAEA TECDOCS [96, 97]. The IAEA also assists States in assessing and improving national programmes through its International Physical Protection Advisory Service (IPPAS).

6.2 General approach to physical protection

The ultimate objective of a physical protection system is to prevent the unauthorised removal of nuclear materials or sabotage of nuclear materials or nuclear facilities. The general approach is to protect against the State's Design Basis Threat through the establishment of a system based on a combination of personnel, hardware, procedures and facility design with due consideration to compatibility with the safety of the facility.

6.2.1 Defining the threat

Definition of the Design Basis Threat (DBT) is a State responsibility and will be carried out by the State appointed agency. Details of the DBT are generally secret, as are the details of measures that are adopted to counter security threats at nuclear and other facilities. Different countries publish different amounts indicating the general nature of the assumed threat for their own national circumstances and facilities. Box 6.1 presents summary information for US civilian facilities, which can be regarded as illustrative.

Box 6.1: The Design Basis Threat at US civilian facilities (from [98])

The US DBT is drawn up by the US Department of Energy in collaboration with other government departments. Prior to September 11th 2001 the scenarios included:

- Attacks by well-trained and dedicated individuals, possibly with military training and skills.
- Attacks involving insider assistance.
- Attacks involving suitable weapons, up to and including hand held automatic weapons, equipped with silencers and having effective long-range accuracy.
- Attacks involving hand carried equipment, including incapacitating agents and explosives.
- Attacks involving a four wheel drive land vehicle to transport attackers and their hand held equipment to the proximity of vital areas.
- An attack involving a four-wheel drive land vehicle bomb and insider assistance.

The DBT was reviewed after September 11th 2001 and a new DBT was introduced in May 2003. Details are not available but the US General Accounting Office (GAO) has indicated that the DBT now includes:

- Attacks by terrorists who are well armed and equipped, trained in paramilitary and guerrilla warfare skills, willing to kill, risk death, commit suicide and capable of attack without warning.
- Attacks by larger groups of terrorists than in the previous DBT.

The DBT is kept under review and updated periodically, as indicated in Box 6.1 for the US DBT. There is also commentary in the public domain on possible modes of attack. Besides those indicated in Box 6.1, some suggested scenarios include:

- Ground-based attacks with heavier ranged weapons, for examples attacks from outside site security perimeters using rockets or light artillery. The US Army has carried out tests of effectiveness of armour piercing missile against a CASTOR storage/transport cask [99].

- Attacks on facilities by air, including crashing of light aircraft loaded with explosives or hijacked commercial aircraft and use of fired or dropped weapons from the air. Consideration of such attacks has attracted high media concern. In practice, however, attacks from the air pose substantially greater logistic and practical problems[22] than ground-based attacks, so that such an attacks would be both more difficult to carry out and at higher risk of prior detection.

- Attacks during transport, including causing accidents by damage to transport infrastructure (e.g. bridges and rail lines), hi-jacking, attack with anti-tank weapons, attack with high explosives or truck bomb, or combinations of any of these.

[22] Noteably, the need to train skilled pilots and the very small target size of a nuclear reactor or store compared, for example, to the World Trade Center or Pentagon buildings that were attacked on September 11th 2001.

6.2.2 Countering the threat

Threats are countered though national and international intelligence gathering, which seeks to identify and counter threats before they materialise, and physical protection systems related to specific targets, such as a nuclear facility. Intelligence gathering is a State responsibility; it is carried out in secret and is outside the scope of this document. Here we discuss only the physical protection systems related to specific targets.

While protection against unauthorised removal or sabotage requires consideration of many common factors, the protection philosophy differs.

- For unauthorised removal, the primary objective is to protect against unauthorised individuals obtaining access to nuclear material and removing it from the facility.

- For sabotage the primary objective is to prevent adversaries from even gaining access to the nuclear material or vital equipment.

While similar concepts are employed for detection and assessment of a potential intrusion, the use of delay features and emergency procedures, including the response force strategy, can be different. For protection against unauthorised removal, the use of penetration delay in barriers securing the material provides time for the guards to call for assistance, and contain or delay the adversaries until the arrival of the response force. For protection against sabotage, the use of delay features or sufficient distance to the target must provide sufficient time for the guards or the response force to interpose themselves between the adversaries and the nuclear material or vital equipment to preclude access to the potential sabotage targets.

To counter a threat of unauthorised removal of nuclear material or the sabotage of nuclear material or nuclear facilities, a protection system should perform the following primary functions:

- deter – detect – assess – delay – respond.

Deterrence

Unauthorised removal or sabotage can be prevented in two ways: by deterring adversaries or by defeating them should they attempt to steal nuclear materials or sabotage nuclear material or nuclear facilities. Deterrence is achieved by implementing a physical protection system that adversaries perceive as too difficult to defeat, making the protected nuclear material or facility an unattractive target. This is a primary aim of a physical protection system.

Detection

Detection is the discovery of an attempted or actual intrusion that could have the objective of unauthorised removal or sabotaging nuclear material or equipment, systems or devices in a protected area. Detection can be accomplished by sensors or personal observation, for example by an employee or guard. To be useful, detection needs to be coupled with an assessment of what has been detected, and the system needs to be able to screen or test to rule out false or non-threatening conditions, e.g. was a sensor triggered by an animal or by weather conditions?

Sensors are an important part of a detection system. By activating alarms they provide an indication of an activity that requires assessment. The goal of any detection system is to maximize the probability of detection while minimising the rate of nuisance alarms. This can be accomplished by providing a continuous line of detection using a single sensor technology appropriate for the environmental conditions and terrain, or by using multiple and complementary sensors that function on different technical principles.

A central alarm station (CAS) is required to continuously evaluate detection and assessment information and communicate with guards and the response force. A reliable communications system between the CAS and the guards and the response force is also essential. The CAS should be hardened, i.e., constructed and located in such a manner so as to allow it to continue operating at all times, even when under attack.

Assessment

Assessment is typically aided by closed circuit television (CCTV) coverage of each sensor sector, complemented by visual checks from guards, either static or mobile. In addition to determining the cause of a detection alarm, assessment should provide specific details such as what, who, where, when and how many, in a timely manner. These details help determine the number of guards who should respond and how they should be equipped. This is vital to allow response forces to react in a timely and effective manner.

Delay

Since it is not possible to maintain a sufficient number of guards at all points to provide immediate protection against all types of adversaries, some means of delaying adversaries is required to provide the guards time to react after the intrusion has been detected and to call for assistance.

Delay can be achieved by barriers, including fences, walls, and locks. Delay should slow the adversaries sufficiently to provide time for the guards or the response force to interpose themselves between the adversaries and their target. Delay should be sufficient to prevent adversaries from accomplishing their mission before guards or response forces can intercede and neutralise the adversaries.

Response

As discussed above, guards and/or the response force need to respond more rapidly to prevent sabotage than to prevent unauthorised removal. Exercises should be performed to ensure the timely response of the guards and/or response force during the critical early stages of an intrusion or attack and to establish the effectiveness of such a response. The resulting experience can be used to develop, correct or modify defensive strategies, monitoring, communications and barriers.

Guards and response forces need to survive in order to prevent adversaries from accomplishing their objectives. Many factors contribute to guard and response force capability and survival, including tactical planning, equipment, weapons, plus training and exercises. Drills should be conducted to demonstrate their effectiveness and improve response capabilities. Consideration may be given to the strategic placement of defensive barriers to provide cover for the guards and the response force.

6.2.3 Robust physical protection system

An effective physical protection system has several specific characteristics. Besides being compatible with a facility's safety system, the physical protection system should provide:

- defence in depth;

- minimum consequence of component failure;
- balanced protection; and
- graded protection in accordance with the significance or potential radiological consequences.

Defence in depth means that for adversaries to accomplish their objectives, they should have to circumvent or defeat a number of different protective devices or barriers in sequence. For example, adversaries may have to penetrate two or more separate barriers before gaining access to a vital area. Defence in depth avoids dependency on one barrier or system (that might fail at the critical period) to counter an attack. The effect produced on adversaries by a system that provides defence in depth will be to:

- increase uncertainty about the physical protection system (and thus possibly deter an attack);
- require more extensive preparation prior to attacking the facility (with the associated greater risk of these preparations being discovered before the attack);
- require different techniques and different tools to penetrate barriers; and
- create additional steps that could cause the adversaries to fail or abort their attack.

Minimum consequence of component failure is an important characteristic because a complex system can always experience some component failure. Causes of component failure in a physical protection system may include environmental factors and tampering by adversaries. Contingency plans are needed so that the system can continue to operate effectively in the event of component failure. Redundant equipment that takes over automatically is desirable in some cases, e.g. an emergency power supply that activates automatically should the primary power source fail.

Balanced protection implies that no matter how adversaries attempt to accomplish their objectives, they will encounter effective elements of the physical protection system. For example the building fabric that surrounds a reactor control room may consist of:

- walls, floors, and ceilings constructed of several types of materials;
- doors of several types; equipment hatches in floors and ceilings; and
- heating, ventilating, and air conditioning openings protected with various types of grills.

Complete balance is probably not possible and not necessary. The delay provided by doors, hatches, and grills may be less than that provided by the walls, but can be adequate if they provide sufficient time for the arrival of response forces and successful interdiction. There is no advantage in over designing, for example, by installing a costly armoured door if the wall is relatively less robust. Both the walls and the doors should provide the appropriate level of protection determined by the DBT, the capabilities of the response forces and the time they need to respond effectively.

The objective should be to provide adequate protection against all scenarios identified in the DBT and to maintain a balance with other considerations, such as cost, safety, and structural integrity. For example, security arrangement must not constrain emergency exit and response arrangements needed for safety.

6.3 Physical protection of a shared waste management system

6.3.1 Waste acceptance

The expectation (see section 4.2) is that a European shared waste management system would accept:

- spent nuclear fuel from civilian power reactors;
- high-level waste from spent nuclear fuel reprocessing;
- long-lived low and intermediate level waste.

Spent nuclear fuel from civilian power reactors, i.e. typically enriched up to a few % ^{235}U, is category II material. HLW and long-lived L/ILW are uncategorised (below category III) on account of the very low concentration of fissile elements and unsuitable form.

Some SAPIERR countries possess quantities of highly enriched uranium fuel (HEU) from research reactors. For the most part, this would be returned to the country of manufacture for disposal, but in any case, the material would still be consistent with category II.

Some SAPIERR countries may possess amounts of plutonium and uranium at high enrichment from research programmes, but the amounts for disposal are expected to be insufficient to constitute category I material.

Hence, the shared waste management system is not expected to accept category I material and this could be set as a boundary condition for waste acceptance.

As concluded by the US General Accounting Office (GAO) [100], "studies indicate a low likelihood of widespread harm to human health from terrorist attacks or severe accidents involving spent fuel – either in transit or dry or wet storage. Spent fuel is a heavy, ceramic material that is neither explosive nor volatile and resists easy dispersal. Tests to date on shipping containers and dry storage containers have shown that, while they can be penetrated under terrorist and severe accident scenarios, their construction allows little release of spent fuel, with little harm to human health".

Vitrified high-level waste would offer even less potential for dispersal. Packaged ILW, e.g. in steel enveloped concrete containers, would also offer little scope for dispersal, the most common waste form being metals or resins mixed into cement grouts.

6.3.2 Waste transport

As remarked in the IAEA PPNMNF [30], see section 6.1, transport of nuclear material is probably the operation most vulnerable to an attempted act of unauthorised removal of nuclear material or sabotage, and suggests several measures to counter the threat:

- minimising the total time during which the nuclear material remains in transport;
- minimising the number and duration of nuclear material transfers, i.e. transfer from one conveyance to another, transfer to and from temporary storage and temporary storage while awaiting the arrival of a vehicle, etc.;
- protecting nuclear material during transport and in temporary storage in a manner consistent with the category of that material;
- avoiding the use of regular movement schedules;

- predetermination of the trustworthiness of all individuals involved during transport of nuclear material; and
- limiting advance knowledge of transport information to the minimum number of persons necessary.

In principle, we consider that a maximum transport by rail, see section 4.4, provides both safety and security advantages over road transport. The advantages are related to restricted public access to rail lines, control of movement schedules and predictable movement rates, and better separation of the waste transport from the public especially in the event of an accident or attempted sabotage.

A European shared waste management system could employ all the above measures indicated by the PPNMNF. In particular:

- the proposed maximum use of rail transport would minimise the total time during which the radioactive waste is in transport;
- allowing large shipments, for example transporting several large transport/storage casks at a time in a single train, thus minimising the number of shipments;
- using dedicated trains, that is shipment would be made in trains containing only waste carrying wagons (not mixed with other freight), to minimise time in marshalling yards and allow direct routing;
- European rail routes are sufficiently extensive and developed to allow alternative routing across most of the distance to be traversed, timing of shipments can also be varied;
- predetermination of the trustworthiness of all employees and other individuals with access to information concerning shipments (and sensitive aspects of all shared facilities and operations) will be a basic precept.

Other features that could be incorporated to further enhance security include:

- where a transfer is needed from road to rail transport, a temporary dedicated secure transfer point could be established;
- shipments would be moved with double (front and rear) locomotives, allowing for single locomotive failure;
- wherever possible, routes would avoid centres of population and conurbations – in most countries, freight lines avoid main stations and population centres, although routing often takes freight though industrial districts etc.;
- on train guards are not usual today on rail transfers of spent fuel and are potentially vulnerable, but responses forces need to be available through communication with military, police and fire service forces en route, and possibly, depending on the threat level, by an independently travelling response force;
- a campaign strategy could be developed to collect spent fuel and other wastes in such an order to minimise overall security risks, considering both the risk during transport and the risk at national storage facilities.

The US GAO has suggested that transport risks could be reduced by picking up fuel in an order that would reduce risk, such as moving older, less radioactive fuel first and removing fuel from pools so as to create space allowing reconfiguration of fuel to lower thermal loads [100].

Studies of the sabotage threat, conducted by Sandia National Laboratory for USDOE in 1999 [101], estimated the amounts and characteristics of releases of radioactive materials from truck and rail spent fuel containers subjected to two different types of weapons. The study confirmed the findings of earlier studies that armour-piercing weapons could penetrate shipping containers and release small quantities of radioactive material. The study found that, under a worst-case scenario, the weapon could penetrate a shipping container and release an amount of material equal to about 0.016 of 1 percent (2/10,000th) of the spent fuel in the container as small, respirable particles. These small, respirable particles could become airborne and spread beyond the immediate vicinity of the attack and would be the cause of subsequent dose via inhalation or external exposure to deposited material. By comparison, non-respirable material would be a more localised problem that could be more easily contained and controlled.

Unauthorised removal of material in transport, e.g. to take material to use in a dirty weapon or as a blackmail tool, is not thought to be credible because of the massive nature of the containers, which would delay any attempts to open or remove material to such an extent that interdiction by responses forces would be certain.

6.3.3 Encapsulation facilities and repositories

Encapsulation facilities and repositories present a rather unattractive target for sabotage or unauthorised removal because of the relatively small amounts of accessible (above ground) nuclear material, i.e. spent fuel, present compared to that located at reactors or storage facilities. The spent fuel that has been emplaced in the repository can be regarded as almost immune from attempts at unauthorised removal. Sabotage of the repository infrastructure, e.g. the shafts or access drift, might be possible but this would not pose a nuclear or radiological hazard.

Nevertheless, protection similar to that employed at reactors or national storage facilities could be deployed and would be aimed primarily at deterrence, see section 6.2.2. Counter-measures can be related to facility design, e.g. siting SF handling and shaft heads within secured areas, and general site security. Typical measures as employed at nuclear power stations include security vetting to establish trustworthiness of all employees and measures to prevent or delay unauthorised access, for example:

- double lines of fencing, high intensity lighting and CCTV linked to a permanently-manned security building;
- turnstiles at personnel access points where entry and exit is only possible with a site-specific electronic pass;
- random searches of personnel and vehicles;
- double barriers at vehicle access points and chicanes to prevent the barriers from being rammed at high speed;
- additional barriers within the station to protect sensitive areas, to which only certain personnel will have access.

These measures may not always deter the determined intruder, but are designed to delay access to sensitive areas long enough to mobilise off-site response forces.

6.3.4 Storage in national facilities

Storage in national facilities is not part of the shared waste management system, but the relative security of radioactive waste, especially spent fuel, in storage as opposed to deposited in a repository is very relevant to the assessment of relative security threats.

The safety of storage of commercial spent fuel has been reviewed by the US GAO as part of its study of options related to security of transport and storage in 2003 [100] and by the Board on Radioactive Waste Management (BRWM) of the US National Academies in a specific study on safety and security of storage in 2004 [102].

The BRWM report

The latter report [102], was requested by the US Congress to provide independent scientific and technical advice on the safety and security of commercial spent nuclear fuel storage in the United States, specifically with respect to the following charges:

- Potential safety and security risks of spent nuclear fuel presently stored in cooling pools at commercial nuclear reactor sites.

- Safety and security advantages, if any, of dry cask storage versus wet pool storage at these reactor sites.

- Potential safety and security advantages, if any, of dry cask storage using various single-, dual-, and multi-purpose cask designs.

- The risks of terrorist attacks on these materials and the risk these materials might be used to construct a radiological dispersal device.

The highlights of the unclassified summary report include the following points:

- Successful terrorist attacks on spent fuel pools, though difficult, are possible. If an attack leads to a propagating zirconium cladding fire, it could result in the release of large amounts of radioactive material. It appears to be feasible to reduce the likelihood of a zirconium cladding fire by various means.

- Dry cask storage has inherent security advantages over spent fuel pool storage, but it can only be used to store older spent fuel. There are no large security differences among different storage-cask designs.

- It would be difficult for terrorists to steal enough spent fuel from storage facilities for use in significant radiological dispersal devices (dirty bombs).

Some selected abbreviated statements from the Committee's findings are as follows.

On the terrorist risk

Spent fuel storage facilities cannot be dismissed as targets for terrorist attacks because it is not possible to predict the behaviour and motivations of terrorists, and because of the attractiveness of spent fuel as a terrorist target given the well known public dread of radiation.

While it would be difficult to attack such facilities, attacks by knowledgeable terrorists with access to appropriate technical means are possible. However, an attack that damages a power plant or its spent fuel storage facilities would not necessarily result in the release of any radioactivity to the environment.

The likelihood that terrorists could steal enough spent fuel for use in a significant radiological dispersal device (a dirty bomb) is small. Removal of a spent fuel assembly from the pool or dry cask would prove extremely difficult under almost any terrorist attack scenario. Attempts by a knowledgeable insider(s) to remove single rods and related debris from the pool might prove easier, but the amount of material that could be removed would be small. Moreover, superior materials could be stolen or purchased more easily from other sources.

The committee recommended that security should be reviewed for the protection of spent fuel rods not contained in fuel assemblies from theft by knowledgeable insiders, especially in facilities where individual fuel rods or portions of rods are being stored in pools.

On the vulnerability of spent fuel stored in pools

Pool storage is required at all operating commercial nuclear power plants to cool newly discharged spent fuel. This fuel must be stored in a pool that has an active heat removal system (i.e., water pumps and heat exchangers) for at least one year before being moved to dry storage. Most dry storage systems are licensed to store fuel that has been out of the reactor for at least five years. Although spent fuel younger than five years could be stored in dry casks, the changes required for shielding and heat-removal.

Under some conditions, a terrorist attack that partially or completely drained a spent fuel pool could lead to a propagating zirconium cladding fire and the release of large quantities of radioactive materials to the environment. It appears to be feasible to reduce the likelihood of a zirconium cladding fire following a loss-of-pool-coolant event using readily implemented measures. These include reconfiguring the spent fuel in the pools to more evenly distribute decay-heat loads and development of a response system to mitigate loss-of-pool-coolant events that would be capable of operation even if the pool or overlying building were severely damaged.

There are substantial differences in the designs of spent fuel pools that make them more or less vulnerable to certain types of terrorist attacks, so that assessment is necessarily case specific.

The committee recommended additional analyses to more fully understand the vulnerabilities and consequences of loss-of-pool-coolant events that could lead to a zirconium cladding fire, and appropriate actions to address any significant vulnerabilities that are identified.

US GAO report

The US GAO report [100] also presents information important to assessing the risk of spent fuel storage, which includes the following.

On the vulnerability of spent fuel pools

NRC studies have reported that a risk of widespread harm to human health from spent fuel arises from the remote possibility of a sustained loss of coolant in a spent fuel pool, potentially leading to a fire that would disperse radioactive material across a wide area. A study of this risk released in 2001 [103] found that, though the potential consequences of such a fire could be severe – nearly 200 early fatalities and thousands of latent cancer fatalities- although the likelihood of such a fire is low. NRC noted that several factors combine to make a pool fire unlikely, including the robust design of the pool; the simple nature of the pool support systems; and the long time required to heat up the fuel, which allows time for operators to respond.

To address some of the uncertainties regarding the risks of storing spent fuel in wet storage pools, NRC has carried out some initial evaluations of sabotage attacks on these pools, and has more work planned and ongoing. Following the terrorist attacks of September 11, 2001, NRC commissioned the US Army Corps of Engineers to examine potential effects of sabotage directed at spent fuel pools. The Corps conducted several computer-based analyses of the potential effects of armour-piercing weapons and high explosives on typical spent fuel pools. The analyses found that the penetration of armour-piercing weapons and high explosives could vary considerably, depending, among other things, on the size of the weapon or explosive and the sophistication of the attacker.

On the vulnerability of dry storage casks

Studies by US DOE and the Corps on dry storage containers have generally reached the same conclusion – that the containers could not be penetrated by airplane crashes and would result in no significant release of radiation when attacked with advanced weapons.

Two DOE-sponsored reports found that airplane crashes would not penetrate dry storage containers. Both reports concluded that although airplane crashes could damage the containers but no radioactive material would be released. The analysis showed that the containers would break up the airplane, spreading jet fuel over a wide area, causing the jet fuel to dissipate or burn without affecting the spent fuel in the containers.

Two other studies, performed in 2001 by the Corps, found that the containers would not release significant amounts of radioactive material when attacked by armour-piercing weapons or high explosives. The study examining the effect of armour-piercing weapons found that the penetration to the containers was very limited. The study examining the effects of high explosives found that the explosives would not completely penetrate the container. The study showed extensive exterior damage, but no penetration to the spent fuel.

6.4 Security overview and comparative risks

6.4.1 The risk of attack and sabotage on civil nuclear targets

The UK Sustainable Development Commission recently published a report on safety and security aspects of nuclear power in the context of possible future role of nuclear power [104]. The report noted that there could be several possible motives for a possible terrorist attack on a nuclear power station, which will depend on the group or groups involved, and on their political aims. Although the detail of the threat will depend on the circumstances pertaining at the time, the report identified a number of general motives:

- to cause widespread death and destruction by direct action;
- to acquire nuclear material which could then be used in an explosive device;
- to cause economic damage;
- to gain publicity for the group in question.

The report further noted that nuclear plants might be considered ‘attractive’ targets for a number of reasons, including:

- the potential to cause wide scale economic and social disruption;
- playing on the public fear and anxiety of radioactivity;
- the possibility of causing a ‘spectacular’ event.

As the BRWM has noted [102], since it is not possible to determine the motivations or aims of terrorist or dissident groups in detail, it is not possible to make a quantitative assessment of risk. Given a list of general motives such as given above, however, it is possible to estimate general attractiveness of alternative civil nuclear targets. Further, it is possible to assess general vulnerability of targets given their characteristics. Combining these features it is possible to assess the relative risks of attacks on civil nuclear targets[23]. Table 6.2 makes a general assessment of the relative attractiveness and risk potential of alternative nuclear targets based on assessment against four general aims derived for the above lists. HLW and ILW storage and transport are considered to be less attractive targets in view of lower potential radiological hazard and lower publicity value.

The assessment and comments presented in Table 6.2 are, of course, subjective but based on the broad understanding of facility characteristics such as described in this report. The following trends emerge:

- Against aim 1 – to cause destruction and deaths by direct action – none of the nuclear targets make an objectively attractive target. Nuclear reactors and spent fuel storage pools offer some possibilities, but rather weak ones. The objective potential of spent fuel storage declines with time as both the heat output and inventory of dispersible radionuclides declines.

- Against aim 2 – to cause economic damage and/or social disruption, including effects of public fear – all targets have some potential through a fear and consequent social disruption effect, even if the attack is unsuccessful in releasing any radioactive material. Disabling a power station would be a real economic detriment, but the fear associated with speculation on potential consequences may be a greater and longer-lasting impact.

- Against aim 3 – to acquire nuclear material which could then be used in an explosive device – none of the targets seem attractive, both because of the quality of nuclear material and extreme difficulty of either reaching or escaping with it.

- Against aim 4 – to gain publicity for the group in question, e.g. causing a ‘spectacular’ event – all the targets offer some possibilities. If publicity is the motive, then the nuclear reactor offers most potential in terms of being a very high value target, but attacks on transport could also be attractive both because of the lesser immediate security and because the event would happen in the public domain.

Overall, we conclude that the nuclear targets make rather poor candidates against aims 1 and 3. Considering aim 2, a nuclear reactor or reactor site (which would also include spent fuel pool storage) present the most attractive targets on account of their high economic value and fear related to the perceived potential of radiological hazards arising from a damaged reactor. Considering aim 4, a nuclear reactor or reactor site would still be a prime target, but transport could be an easier target to strike and would have the advantage that the event takes place in the public domain.

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[23] We do not consider risks to nuclear weapons establishments or materials, or civil reprocessing facilities, since none of the SAPIERR countries possess such facilities or materials.

Attractiveness or practical potential of nuclear targets to meet possible aims of terrorist or activist groups				
Nuclear reactor	Spent fuel pool storage	Spent fuel cask storage	Spent fuel transport	Encapsulation and repository
Aim 1: to cause destruction and deaths by direct action				
Theoretically possible from uncontrolled fission reaction. Practically impossible due to reactor passive and automatic safety systems	Possibility of widespread contamination following pool draining and zirconium alloy fire. Unlikely and recoverable before overheating.	Negligible potential. Robust containment and limited potential for dispersal.	Negligible potential. Robust containment and limited potential for dispersal.	Negligible potential. Only small amounts of SF uncontained at any one time.
Aim 2: to cause economic damage and/or social disruption, including effects of public fear				
Disruption of power generation possible through damage to reactor and/or control systems. Possible local evacuation and severe fear impacts.	Evacuation and possible fear impacts if facility is damaged.	No objective impact, but possible fear impacts locally.	Minor objective impacts related to disruption of transport routes. Fear impacts locally.	No objective impact, but possible fear impacts locally.
Aim 3: to acquire nuclear material which could then be used in an explosive device				
Radiation hazards make it practically impossible.	Difficult. Insider removal of single rods or part rods has been suggested.	Near impossible. Specialist heavy equipment needed and would take considerable time to take SF from casks.	Near impossible. Specialist heavy equipment needed and would take considerable time to take SF from casks.	Negligible potential. Only small amounts of SF uncontained at any one time.
Aim 4: to gain publicity for the group in question, e.g. causing a ‘spectacular’ event				
High value target. Perceived very high potential consequence. High media value.	Perceived high potential consequence.	Low potential consequence.	Low potential consequence, but high public concern.	Low potential consequence.

Table 6.2: Assessment of nuclear targets to meet the possible aims of terrorist or activist groups

6.4.2 Non-proliferation and nuclear safeguards

The previous sections of this chapter have focused on the threat from non-state actors (terrorists, saboteurs and activists) who might seek to damage nuclear facilities or steal nuclear material. There is also the threat from states or their agents that might seek to divert nuclear materials from their civil nuclear programmes to military use. This threat is minimised by nuclear safeguards established under the Treaty on the Non-proliferation of Nuclear Weapons as described in section 3.4.

We consider that it is not appropriate to make an assessment of security risks of state-sponsored diversion of nuclear materials because:

- nuclear safeguards are equally applicable to shared or national nuclear activities and enforced under the same internationally-supervised arrangements;
- any assessment would need to make judgements about long-term political stability and intentions that are both speculative and political.

In principle, however, a shared repository programme does offer security advantages from non-proliferation and nuclear safeguards perspectives for at least two reasons:

A reduction of the number of sites at which nuclear material is held, so that nuclear safeguards effort can be focussed on those fewer locations and facilities. In particular, nuclear material will be removed from the territories of countries that may find the long-term supervision of small amounts of nuclear materials a burden, and also the eventual number of final repositories will be reduced.

The prospect of more rapid progress of smaller programmes towards emplacement of their nuclear materials in an underground repository, so that the intrinsic security advantages of geological disposal are realised sooner, see section 6.4.4.

6.4.3 Comparative risks for shared or national management systems

The qualitative assessment of security in the section 6.4.1 has some parallels with the quantitative assessment of collective doses in sections 5.6.4 and 5.6.5. That is: (1) the assessed risks from the stages of spent fuel and radioactive waste management are less than those related to the nuclear reactor operations that created the spent fuel and radioactive waste; and (2) the long-term storage of spent fuel especially in pool storage presents a long-term target and hence risk. Moreover, security risks become difficult to estimate into the future, given uncertain motives and capabilities of terrorist or activists and uncertainty over security measures, especially in countries in which a nuclear industry is not sustained.

Potential security benefits (both to counter proliferation and improve physical protection) have been one of the main motives for consideration of multilateral approaches in general, see [105], and development of multinational repositories in particular, see [95].

The IAEA TECDOC on the development of multinational repositories [95] notes that:

- the concept of multinational repositories offers the opportunity of safe and secure radioactive waste disposal to countries that are not able for various reasons to implement a national repository project in a timely fashion; and

- that the expected global security and safety benefits provide major arguments for supporting the concept of multinational repositories and for encouraging potential host countries to offer their cooperation to interested partner countries.

It is important to note that the improvements in safety and security that are expected are at a global scale. It is not intended to imply that a multinational repository will be safer or more secure than a properly implemented national repository. The global benefit results from making a proper disposal facility accessible also to countries that may not be in a position to implement a state of the art national repository.

Multinational repositories may also increase some security risks. Primarily, a multinational repository will involve transport between the partner and host country over longer distances. This could result in increased risks of sabotage against nuclear material during transport. Transport is, however, a transitory process and there are strategies by which the risks can be reduced, see section 6.3.2.

6.4.4 General security advantages of geological disposal

It is recognised that geological disposal offers general security advantages over surface storage with respect to both physical protection and non-proliferation, e.g. see [106] and [95].

Compared with surface storage of nuclear materials, emplacement in a geological repository provides a higher level of security because of the lower accessibility of the waste material. The emplacement of SF deep underground inside a facility that is monitored, with numerous engineered and administrative controls, can enhance both physical security and safeguards relative to most surface storage facilities. After closure, the risk of clandestine human intrusion is highly unlikely because of the long time required for the assembly and operation of the mining and drilling equipment necessary for entry, and because of the detectable signals and indicators associated with those activities.

Conditions for security of a multinational repository relate to measures necessary to guarantee the non-proliferation and physical protection. Prior to the terrorist attacks on the USA in September 2001, proliferation of weapons-grade nuclear material was arguably the primary security concern associated with nuclear facilities. Since then, however, the threat of terrorist attacks against nuclear facilities that could result in the release of radioactive debris into the atmosphere and the unauthorised removal of radioactive material that could be used in radioactive dispersal devices has greatly expanded security concerns and emphasised the need for physical protection as described in this report.

The IAEA notes [95] that multinational repositories could make the security and safeguards benefits of geological disposal available to more countries and could therefore enhance global security. Provided that the host country, most likely in agreement with the partner countries, takes the appropriate measures, a highly transparent, safeguarded and well-protected multinational repository could greatly reduce the risk of proliferation through theft or diversion of the material in the repository. A multinational repository could be specifically sited, designed and constructed to create high levels of security.

Thus, a conclusion similar to that for safety in section 5.7.4 can be reached. That is, if the combined efforts of several countries give better prospect for joint realisation of a disposal project at an earlier time, then a security benefit arises, primarily by reducing the amounts of nuclear material contained in surface storage facilities and the length of time for which it is so stored.

7 Conclusions

This chapter presents a summary of conclusions that are supported and discussed in the preceding chapters, plus final remarks on common factors that underpin safety and security.

7.1 The SAPIERR projects and this report

The basic concept within both SAPIERR I and SAPIERR II is that of one or more geological repositories developed in collaboration by two or more European countries to accept spent nuclear fuel, vitrified high-level waste and other long-lived radioactive waste from those countries.

The SAPIERR II project (Strategic Action Plan for Implementation of Regional European Repositories) examines in detail issues that directly influence the practicability and acceptability of such facilities. To achieve this it is necessary this is to consider the complete chain of activities and facilities that would be needed take radioactive waste from storage facilities at nuclear power plants, or from centralised national storage facilities, to final disposal in one or more shared deep geological repositories.

This report is produced under Work Package 4 of SAPIERR II, the aim of which is to make an outline examination of the safety and security aspects of implementing one or two regional repositories within the European Union, relative to a larger number of national repositories.

Rigorous assurances of safety and security are essential requirements for all such activities and facilities, whether carried out nationally or on a shared basis. Experience indicates that with proper attention to siting, design, quality of implementation, monitoring and control, and provided that the necessary financial and technical resources are committed, the feasibility of achieving the required standards of safety and security is not in question.

The emphasis in this report, therefore, is:

- to survey the safety and security standards that would apply to a multi-national radioactive waste management system leading to final disposal within one or more shared repositories in the EU;
- to confirm the methods and techniques that are available to assure safe and secure accomplishment of all the necessary waste management steps, and to indicate their performance;
- to make simple generic comparisons and assessments of safety and security aspects of implementing such a system, compared to that of implementing a number of national systems.

The focus is on nuclear safety (i.e. radiological safety) and nuclear security. This is because it is the nuclear and radiological aspects that are the special and defining aspects of the proposal for shared geological repositories in the EU. They are, therefore, the most important aspects to consider at this conceptual stage.

7.2 Safety and security standards

Safety and security are the highest-level goals for radioactive waste and spent fuel management, and must be assured both during waste management operations and, in the long term, after disposal is complete.

Chapter 3 outlines the boundary conditions that must be met by any national or multinational system for the long-term management of radioactive waste in the EU. This includes obligations under international treaties and agreements and the internationally-developed objectives and principles related to safety, security and nuclear safeguards. Societal acceptance, political decisions and costs will also be important boundary conditions, but are not discussed in this report.

A shared radioactive waste management system would be subject to all the requirements of the relevant treaties and agreements, through the obligations of the states in which radioactive waste originates, through which waste is transported, and in which the shared facilities are sited.

Common international guidance and EU laws will apply to any national or international radioactive waste management system developed within the EU. High standards of safety and security will be demanded by society and national governments, sought by the developer and enforced by regulatory bodies. Hence any radioactive waste management system developed in the EU will be safe and secure – where this means as safe and secure as it can reasonably be made (applying best practical means and ensuring doses and risks are ALARA) and in compliance national laws and regulations.

The shared facilities and transport shipments would have to meet the national legal and regulatory requirements of the countries in which they are implemented, under the oversight of the national regulatory safety and security bodies. These bodies, most likely supported by equivalent bodies in partner countries, can reasonably demand that a shared facility meets standards of practice for design, assessment and implementation at least as good as established in other EU countries.

It can be expected that, at the time of beginning implementation of shared European facilities, several high-quality long-term interim stores and geological disposal facilities will already be operating or under construction in the EU. The design, assessment and implementation of regional facilities will take advantage of this experience. It is also likely that the EU may offer specific guidance and/or oversight for a shared project.

7.3 Assessment of safety

Chapter 5 discusses nuclear and radiological safety aspects of radioactive waste management from waste acceptance to disposal. This includes discussion of safety standards, waste and waste package acceptance, transport safety, operational safety of facilities, and repository post-closure safety. Proportionately, most attention is directed at safety related to spent nuclear fuel, which is radiologically the dominant waste form.

The chapters confirm that radiological safety is achievable for all steps required within a European shared waste management system. This has been demonstrated in practice for the steps of radioactive waste handling, transport and storage, including for spent nuclear fuel. The step of sealing of SF/HLW into disposal containers has not been demonstrated, but appropriate technologies and have been developed and tested, and the radiological protection measures are the same as those already in use for handling SF/HLW. ILW is routinely packaged for storage and disposal in many countries. Radiological safety assessments and practical experience shows that the necessary steps can be safety accomplished in accord with international guidance and in compliance with national laws and regulations.

Licensing and operation of a deep geological repository for SF, HLW or long-lived ILW has not been demonstrated, but several countries are working towards that goal, and there is every reason to believe that such facilities will be brought into operation within the next two decades.

The final section of Chapter 5 provides a safety overview and presents indicative estimates of radiological impacts for a shared waste management system and equivalent capacity national systems. The assessments show there is little difference between calculated radiological impacts for a large or small inventory shared European spent fuel management system and several national systems with equivalent capacities. The most important quantitative difference, or potential dose reduction, arises from the assumption that timely development of a shared repository would reduce the average time that spent fuel is stored at national facilities, especially pool storage facilities. Even so, the calculated collective dose reductions (to workers and to members of the public) are only about 1/1000th of the collective doses from the reactor operations that produced the spent fuel.

In these dose comparisons, post-closure radiological impacts do not figure because, for an appropriately sited and well-designed geological repository, no releases to the environment are expected until many thousands of years after closure. We believe this is the correct perspective on foreseeable radiological impacts from spent fuel management, and correctly assesses the relative radiological impacts of shared versus national disposal.

On the other hand, the specific aim of geological disposal is to provide assurance of safety over very long times, up to the order of several millions of years. Thus, even if a tangible radiological benefit cannot be shown, it is worth considering whether a shared system offers any advantage in this respect. The following factors are identified as most relevant.

- The greater choice of geological situations and sites available over several countries could provide better opportunities for finding a geologically “better” site than might be found in some smaller countries. It should be emphasised, however, that it will not be the intention to find a “best geological site”. Given several sites that are estimated to provide suitable conditions, the choice between sites is likely to rest on other factors. We thus conclude that a shared repository project could give better options for geological site selection, but a long-term safety advantage may not result.

- A larger pool of financial and human resources will give a stronger basis for implementation. A well-focussed, co-operative effort from several countries can lead to a fuller and more critical consideration of safety and technical issues at each step, and thus a better quality of implementation may be achieved. This is a general argument in favour of a shared repository project that applies to all aspects of technical implementation, safety and security, not just long-term safety.

- Greater international scrutiny with assured regulatory oversight through the multinational agreements. The increased multinational oversight and peer review could improve implementation, but the benefits may be related to confidence and transparency rather than actual long-term performance.

A final factor is that the combined efforts of several countries may give better prospect for joint realisation of a project at an earlier time than if national projects proceed independently. This can provide a tangible safety benefit due to a reduction in the average time that spent fuel is stored at national facilities, and also a less quantifiable but important safety benefit of less chance that disposal will be indefinitely delayed in any country.

7.4 Assessment of security

Chapter 6 discusses security aspects of a radioactive waste management system from waste acceptance to disposal. This includes discussion of nuclear security standards, defining and countering security threats, and physical protection systems in general terms. Security aspects of a shared waste management system and its stages are then discussed, and conclusions are drawn on the security of a shared system compared to a case of several smaller national systems.

The emphasis is on physical protection systems, the prime objective of which is to establish conditions that will minimise the possibilities for unauthorised removal of nuclear material and/or for sabotage of nuclear material or facilities. Non-proliferation and nuclear safeguards are not discussed in detail, since the controls are equally applicable to shared or national nuclear activities and under the same internationally-supervised arrangements. The reduction of number of locations at which nuclear materials will be stored is, however, a positive feature in favour of a shared repository project.

The principles and implementation requirements for the physical protection of nuclear material and nuclear facilities are set down in international guidance. To counter a threat of unauthorised removal of nuclear material or sabotage, a protection system is developed to deter, detect, assess, delay and respond to possible threats and incursions. Besides being compatible with a facility's safety system, the physical protection system should provide defence in depth, minimum consequence of component failure, balanced protection and graded protection in accordance with the significance or potential radiological consequences.

The chapter confirms that nuclear security is achievable for all steps required within a European shared waste management system. This has been demonstrated in practice for both fixed facilities (i.e. nuclear reactors, storage and encapsulation facilities, and repositories) and transport, including for spent nuclear fuel. Security assessments and practical experience shows that the necessary steps can be securely accomplished in accord with international guidance and in compliance with national laws and regulations.

The final section of Chapter 6 presents a qualitative assessment of alternative nuclear targets against four generic terrorist aims.

- Against aim 1 – to cause destruction and deaths by direct action – none of the nuclear targets make an objectively attractive target. Nuclear reactors and spent fuel storage pools offer some possibilities, but rather weak ones. The objective potential of spent fuel storage declines with time as both the heat output and inventory of dispersible radionuclides declines.

- Against aim 2 – to cause economic damage and/or social disruption, including effects of public fear – all targets have some potential through a fear and consequent social disruption effect, even if the attack is unsuccessful in releasing radioactive material. Disabling a power station would be a real economic detriment, but the fear associated with speculation on potential consequences may be a greater and longer-lasting impact.

- Against aim 3 – to acquire nuclear material which could then be used in an explosive device – none of the targets seem attractive, both because of the quality of nuclear material and extreme difficulty of either reaching or escaping with it.

- Against aim 4 – to gain publicity for the group in question, e.g. causing a ‘spectacular’ event – all the targets offer some possibilities. The nuclear reactor offers most potential in terms of being a very high value target, but attacks on transport could also be attractive both because of the lesser immediate security and because the event would happen in the public domain.

The qualitative assessment of security thus has parallels with the quantitative assessment of collective doses in Chapter 5. That is: (1) the security risks from the stages of spent fuel and radioactive waste management are probably less than those related to the nuclear reactor operations that create the spent fuel and radioactive waste; and (2) the long-term storage of spent fuel especially in pool storage presents a long-term target and hence security risk. Moreover, security risks become difficult to estimate into the future, given uncertain motives and capabilities of terrorist or activist groups and uncertainty over security measures, especially in countries in which a nuclear industry is not sustained.

We are thus able to support the statement of IAEA group on developing multinational repositories, that such repositories could make the security and safeguards benefits of geological disposal available to more countries and could therefore enhance global security. Provided that the host country, most likely in agreement with the partner countries, takes the appropriate measures, a highly transparent, safeguarded and well-protected multinational repository could greatly reduce the risk of proliferation through theft or diversion of the material in the repository. A multinational repository could be specifically sited, designed and constructed to create high levels of security.

7.5 Summary of safety and security assessments

High levels of safety and security will be applied to the management and final disposal of radioactive waste and spent nuclear fuel in both national and shared projects.

This report shows that the required safety and security standards are achievable for all required steps and confirms that a shared project presents no technical issues that will not have to be overcome in national projects.

The assessed radiological safety of a notional shared waste management system shows a small collective dose reduction relative to national waste management systems of the same capacity. This arises from an assumption that timely development of a shared repository would reduce the average time that spent fuel is stored at national facilities. The calculated collective dose reductions (to workers and to members of the public) are only about 1/1000th of the collective doses from the reactor operations that produced the waste.

A qualitative assessment of security indicates the security risks of shared or national waste management systems are very similar, and in both cases less than the security risk posed by operating nuclear reactors. Transport of spent fuel over the longer distances required within a shared waste management system poses an added risk, but there are strategies by which the risk can be reduced.

A general benefit of the development of a shared waste management system is that a well-focussed, co-operative effort from several countries can lead to a fuller and more critical consideration of safety, security and other issues at each step, and thus a better quality of implementation may be achieved.

We also consider that the combined efforts of several countries may give better prospect for joint realisation of a project at an earlier time than if national projects proceed independently. This presents a small a tangible benefit due to a reduction in the average time that spent fuel is stored at national facilities, and also a less quantifiable benefit of less chance that disposal will be indefinitely delayed in any country. We fully support, however, the view of the IAEA group on developing multinational radioactive waste repositories that:

> *"the improvements in safety and security that are expected are at a global scale. It is not intended to imply that a multinational repository will be safer or more secure than a properly implemented national repository. The global benefit results from making a proper disposal facility accessible also to countries that may not be in a position to implement a state of the art national repository."*

7.6 Common issues that underpin safety and security

Finally, we note that there are common issues that underpin safety and security of any radioactive waste management system, that are organisational and strategic. These include:

- Responsibility – practical and legal allocation responsibilities, responsible actors, development of safety and security cultures;
- Quality of planning & design – for waste conditioning, transport, storage and disposal facilities;
- Quality of implementation – site selection, engineering, protection procedures and controls, monitoring and assessment
- Effective oversight & regulation – integrated planning, system of requirements, system of compliance, legal sanctions;
- Secured resources – for implementation and for regulation;
- Economic and cultural stability – continued financial and human resources, consistent perceptions and values;
- Inter-organisational (& inter-national) understanding and co-operation – shared goals, effective communication, perceived equity of transactions (now and into the future);
- Timescales – timely conditioning, transport to stores and disposal is crucial to managing environmental risks, safety risks and security risks.

These issues must be addressed in both national and multi-national waste management systems. We consider, however, that a well-focussed, co-operative effort from several EU countries, with the support of the EU, may give better conditions to promote the necessary organisational conditions and qualities than might be achieved in some smaller EU countries working on their own. It will also be more efficient in terms of human and technical resources.

8 References

1 International Atomic Energy Agency, Developing multinational radioactive waste repositories: Infrastructural framework and scenarios of cooperation, IAEA-TECDOC-1413, IAEA Vienna, October 2004.

2 Stefula, V., SAPIERR Support Action: Pilot Initiative for European Regional Repositories. Final Report Deliverable D-7. European Commission, Community Research series, 2006.

3 Boutellier Ch. and McCombie Ch., Technical Report on Legal Aspects (Deliverable 2). Support Action: Pilot Initiative for European Regional Repositories (SAPIERR). ARIUS, Baden, November 2004.

4 Chapman N., McCombie Ch., Stefula V., Possible options and scenarios of regional disposal and future RTD recommendations (Deliverable 3). Support Action: Pilot Initiative for European Regional Repositories (SAPIERR). ARIUS, Baden, November 2005.

5 Stefula V., Inventory, SAPIERR (Deliverable 1), Support Action: Pilot Initiative for European Regional Repositories (SAPIERR), September 2004.

6 International Atomic Energy Agency, Classification of Radioactive Waste, Safety Series No. 111-G-1.1, 1994.

7 International Atomic Energy Agency, IAEA Safety Glossary: Terminology used in nuclear, radiation, radioactive waste and transport safety, Version 2.0, September 2006.

8 International Atomic Energy Agency, Geological Disposal of Radioactive Waste, Safety Requirements No. WS-R-4, 2005.

9 Hedin A, Spent nuclear fuel – How dangerous is it? A report from the project "Description of Risk", SKB Technical Report 97-13, March 1997.

10 Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management, International Atomic Energy Agency Information Circular INFCIRC/546, 24 December 1997. http://www-ns.iaea.org/conventions/waste-jointconvention.htm

11 National Reports to the second meeting of the contracting parties to the Joint Convention on the safety of spent fuel management and the safety of radioactive waste management, May 2006. http://www-ns.iaea.org/conventions/waste-jointconvention.htm

12 IAEA, The Agency's Safeguards System (1965, as Provisionally Extended in 1966 and 1968) INFCIRC/66/Rev.2, 16 September 1968. http://www.iaea.org/Publications/Documents/Infcircs/Others/inf66r2.shtml

13 The Convention on the Physical Protection of Nuclear Material, International Atomic Energy Agency Information Circular INFCIRC/274/Rev.1, May 1980. http://www.iaea.org/Publications/Documents/Infcircs/Others/inf274r1.shtml

14 Nuclear Security - Measures to Protect Against Nuclear Terrorism, Amendment to the Convention on the Physical Protection of Nuclear Material, International Atomic Energy Agency GOV/INF/2005/10-GC(49)/INF/6, 6 September 2005.

15 Kirton J. and Sunderland L., The G8 Summit Communiqués on Nuclear Non-Proliferation, 1975-2005, G8 Research Group, November 2005. http://www.g7.utoronto.ca/references/nonproliferation.pdf

16 Amended proposal for a Council Directive (Euratom) laying down basic obligations and general principles on the safety of nuclear installations, 2003/0021 (CNS), in COM(2004) 526 final, Brussels, 8.9.2004.

17 European Commission, Council Directive 96/29/Euratom of May 13, 1996 laying down the basic safety standards for the protection of the health of workers and the general public against the danger arising from ionizing radiation, Official Journal of the European Communities No. L 159 of June 29, 1996. http://ec.europa.eu/energy/nuclear/radioprotection/doc/legislation/9629_en.pdf.

18 International Commission on Radiological Protection, 1990 Recommendations of the ICRP, ICRP Publication 60, Pergamon Press, Oxford and New York, 1991.

19 International Commission on Radiological Protection, Recommendations of the ICRP, Publication 103, 2007.

20 FAO, IAEA, ILO, OECD/NEA, PAHO, WHO, International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources, International Atomic Energy Agency Safety Series No. 115, Vienna, 1996.

21 International Atomic Energy Agency, Fundamental Safety Principles, Safety Fundamentals No. SF-1, Vienna, 2006.

22 International Commission on Radiological Protection, Radiological Protection Policy for the Disposal of Radioactive Waste, ICRP Publication 77, Pergamon Press, Oxford and New York, 1997.

23 International Commission on Radiological Protection, Radiation Protection Recommendations as Applied to the Disposal of Long Lived Solid Radioactive Waste, ICRP Publication 81, Pergamon Press, Oxford and New York, 2000.

24 International Atomic Energy Agency, Regulations for the Safe Transport of Radioactive Material, 2005 Edition, IAEA Safety Standards Series No. TS-R-1, IAEA, Vienna, 2005. http://www-pub.iaea.org/MTCD/publications/PDF/Pub1225_web.pdf

25 International Atomic Energy Agency, Predisposal Management of Radioactive Waste, Including Decommissioning, IAEA Safety Standards Series No. WS-R-2, IAEA, Vienna, 2000.

26 IAEA and OECD/NEA, Geological Disposal of Radioactive Waste, Safety Requirements No. WS-R-4, 2005. http://www-pub.iaea.org/MTCD/publications/PDF/Pub1231_web.pdf

27 IAEA, Storage of radioactive waste, Safety Guide SW-G-6.1, Vienna, 2006. http://www-pub.iaea.org/MTCD/publications/PDF/Pub1254_web.pdf

28 IAEA, Predisposal management of high level radioactive waste, Safety Guide WS-G-2.6, Vienna, 2003. http://www-pub.iaea.org/MTCD/publications/PDF/Pub1151_web.pdf.

29 IAEA, Design of Spent Fuel Storage Facilities, Safety Series No. 116, IAEA, Vienna, 1994. http://www-pub.iaea.org/MTCD/publications/PDF/Pub976e_web.pdf

30 IAEA, The physical protection of nuclear material and nuclear facilities, INFCIRC/225/Rev 4 (corrected) June 1999, http://www.iaea.org/Publications/Documents/Infcircs/1999/infcirc225r4c/rev4_content.html

31 IAEA, The Agency's Safeguards System (1965, as Provisionally Extended in 1966 and 1968) INFCIRC/66/Rev.2, 16 September 1968. http://www.iaea.org/Publications/Documents/Infcircs/Others/inf66r2.shtml

32 Chapman N., McCombie Ch. and Richardson P., Work Package 3, Economic Aspects of Regional Repositories, SAPIERR II report, April 2008.

33 Sumerling T., The International RD&D Basis for Geological Repositories, Safety Assessment Management Ltd. Report to United Kingdom Nirex Limited, March 2006. http://www.sam-ltd.com/publications_authored.html

34 Chapman N., McCombie Ch. and Richardson P., Work Package 3, Economic Aspects of Regional Repositories, SAPIERR II report, April 2008.

35 FAO, IAEA, ILO, OECD/NEA, PAHO, UNOCHA, WHO, Preparedness and Response for a Nuclear or Radiological Emergency, IAEA Safety Standards Series No. GS-R-2, IAEA, Vienna, 2002.

36 Environment Agency of England and Wales and Environment and Heritage Service for Northern Ireland, Deep Geological Disposal Facilities on Land for Solid Radioactive Wastes: Guidance on Requirements for Authorisation, Daft for Public Consultation, 15 May 2008.

37 Ferch R., A comparison of national dose and risk criteria for deep disposal of long-lived radioactive waste, pp89-98 in Safety Cases for Deep Geological Disposal of Radioactive Waste: Where do we Stand? Symposium Proceedings, Paris, 23-25 January 2007, OECD, NEA No. 6319, 2008.

38 Nuclear Energy Agency, Regulating the Long-term Safety of Geological Disposal: Towards a Common Understanding of the Main Objectives and Bases of Safety Criteria, OECD Nuclear Energy Agency NEA No. 6182, 2007.

39 United Kingdom Nirex Limited, Generic Repository Studies, Generic Operational Safety Assessment, Part 1 – Operational Safety Report, Nirex Report no. N/079, July 2003.

40 SAPIERR ii WP 2 report .. legal etc.

41 IAEA, Advisory Material for the IAEA Regulations for the Safe Transport of Radioactive Material (1996 Edition)", IAEA Safety Standards Series No. TS-G-1.1 (ST-2).

42 IAEA, Planning and Preparing for Emergency Response to Transport Accidents Involving Radioactive Material, IAEA Safety Standards Series No. TS-G-1.2 (ST-3). "

43 IAEA, Compliance Assurance for the Safe Transport of Radioactive Material", IAEA Safety Standards Series No. TS-G-1.4.

44 IAEA, Quality Assurance for the Safe Transport of Radioactive Material", IAEA Safety Standards Series No. TS-G-1.3.

45 IAEA, Operation and Maintenance of Spent Fuel Storage and Transportation Casks/Containers, IAEA TECDOC-1532, January 2007.

46 Haire M.J. et al., Characteristics of next-generation spent nuclar fuel (SNF) transport and storage casks, Americas Nuclear Energy Symposium, Miami Beach, Florida, October, 2004.

47 United Kingdom Nirex Limited, Generic Repository Studies - Generic Waste Package Specification, Volume 1: Specification, Nirex Report N/104, 2005.

48 United Kingdom Nirex Limited, The viability of a phased geological repository concept for the long-term management of the UK's radioactive waste, Nirex Report no. N/122, November 2005.

49 Garrick J.B, The Current Status, Safety, and Transportation of Spent Nuclear Fuel, National Academy of Engineering Website, http://www.nae.edu/nae/bridgecom.nsf/weblinks/MKEZ-5RXRGL?OpenDocument

50 Jefferson, R.M., and H.R. Yoshimura, Crash Testing of Nuclear Fuel Shipping Containers, Sandia National Laboratories report SAND77-1462C. Albuquerque, N.M., 1977.

51 Droste B., Probst U. and Heller W., Impact of an exploding LPG rail tank car onto a CASTOR spent fuel cask, Radioactive Waste Management and Transport, Vol. 10, No. 4, pp. 231–240 (1999).

52 Bentele W. and T Kinzelmann T., Experience with the loading and transport of fuel assembly transport casks, including CASTOR casks, and the radiation exposure of personnel. J. Radiol. Prot. 19, 353-359, December 1999.

53 Tunaboylu K., Playfair A. and Mariapillai N., Waste Transport and Public Safety, Pangea Technical Report PTR-01-03, July 2001.

54 United Kingdom Nirex Limited, Generic Transport Safety Assessment, Volume 1 - Main Report, Nirex Report no. N/078, July 2003.

55 Grahn P.H. and Skogsberg M., The Experience of Interim Storage of Spent Nuclear Fuel and Disposal of Radioactive Waste in Sweden, Swedish Nuclear Fuel and Waste Management Co, SKB. ENC 2005.

56 STUK, Operational Safety of a Disposal Facility for Spent Nuclear Fuel, Guide YVL 8.5 / 23 December 2002.
http://www.edilex.fi/content/stuklex/en/lainsaadanto/saannosto/YVL8-5/

57 United Kingdom Nirex Limited, Generic Operational Safety Assessment, Part 1 - Operational Safety Report, Nirex Report no. N/079, July 2003.

58 SKB, Long-Term Safety for KBS-3 Repositories at Forsmark and Laxemar – a First Evaluation (Main Report of the SR-Can Project). SKB Report TR-06-09, 2006.

59 Nagra, Project Opalinus Clay. Safety Report. Disposal Feasibility for Spent fuel, Vitrified High-level Waste and Long-lived Intermediate-level Waste (Entsorgungsnachweis), Technical Report 02-05, December 2002.

60 SKB, Deep Repository for Spent Nuclear Fuel: SR97 – Post-closure safety, Volume 1, SKB Technical Report TR-99-06, November 1999.

61 Graf R. and Filbert W., Disposal of spent fuel from German nuclear power plants – paper work or technology? TOPSEAL 2006, Transactions International Topical Meeting, Olkiluoto Finland, September 2006.

62 SKB, Äspö Hard Rock Laboratory, Annual Report 2003, SKB Technical Report TR-04-10, SKB, Stockholm, 2004.

63 Bel J., Van Cotthem A., De Bock C., Construction, operation and closure of the Belgian repository for long-lived radioactive waste. ICEM'05: The 10th International Conference on Environmental Remediation and Radioactive Waste Management. September 4-8, 2005, Glasgow, Scotland. Reference ICEM05-1339, 2005.

64 EC 2005, ESDRED: Engineering Studies and Demonstration of Repository Designs, Technical Abstract. ESDRED/F16W-CT-2003-508851, 2005.
www.Andra.fr/IMG/pdf/ESDRED_Abstract-5.pdf also see www.esdred.info/

65 Verhouf E., personal communication, 14 July 2008.

66 ANDRA, How can the safety of a repository be guaranteed during operation?
http://www.andra.fr/interne.php3?id_rubrique=153

67 SKB, Plan 2003: Costs for Management of the Radioactive Waste Products from Nuclear Power Production, SKB Report TR-03-11, 2003.

68 Baldwin T., Chapman C. and Neall F., Geological Disposal Options for High-level Waste and Spent Fuel, Report for the UK Nuclear Decommissioning Authority, January 2008.

69 Nuclear Energy Agency, Post-closure Safety Case for Geological Repositories: Nature and Purpose, OECD/NEA No. 3679, OECD, Paris, 2004.

70 Nuclear Energy Agency, Confidence in the Long Term Safety of Deep Geological Repositories: Its Communication and Development, OECD/NEA, Paris, 1999.

71 International Atomic Energy Agency, Geological Disposal of Radioactive Waste, Safety Guide, in preparation.

72 Nuclear Energy Agency, International Atomic Energy Agency and Commission of the European Communities. Can long-term safety be evaluated? An International Collective Opinion, OECD, Paris, 1991.

73 Nuclear Energy Agency, The Environmental and Ethical Basis of Geological Disposal. A Collective Opinion of the NEA Radioactive Waste Management Committee, OECD, Paris, 1995.

74 Nuclear Energy Agency, Progress Towards Geologic Disposal of Radioactive Waste: Where Do We Stand? An International Assessment. OECD, Paris, 1999.

75 US National Academies of Science. Disposition of High-level Waste and Spent Nuclear Fuel: The Continuing Societal and Technical Challenges. National Academies Press, Washington, 2001.

76 International Association for Environmentally Safe Disposal of Radioactive Materials (EDRAM), Information Leaflet, 2004.

77 SKB. Appendix A: SKBís Master Plan, in RD&D Programme 2004, Programme for research, development and demonstration of methods for the management and disposal of nuclear waste, including social science research, SKB Technical Report TR-04-21, Stockholm, Sweden, 2004.

78 EKRA, Disposal Concepts for Radioactive Waste ñ Final Report, Report of the Expert Group on Disposal Concepts for Radioactive Waste (EKRA), Federal Office of Energy, Bern Switzerland, January 2000.

79 US National Academies of Science. One Step at a Time: The Staged Development of Geologic Repositories for High-Level Radioactive Waste. Committee on Principles and Operational Strategies for Staged Repository Systems, National Academies Press, Washington, 2003.

80 International Atomic Energy Agency. Retrievability of High Level Waste and Spent Nuclear Fuel, Proceedings of an International Seminar organised by KASAM in co-operation with IAEA, Sweden 1999, published as IAEA-TECDOC-1187, Vienna, 2000.

81 Nuclear Energy Agency. Reversibility and Retrievability in Geologic Disposal of Radioactive Waste: Reflections at an International Level, OECD, Paris, 2001.

82 Sumerling T and Smith P, Disposal of Nuclear Fuel Waste - Geological Alternatives, Designs, Implementation, and Long-term Safety, Interdisciplinary Science Reviews Vol. 23, No.3, The Institute of Materials, 1998.

83 International Atomic Energy Agency, Scientific and Technical Basis for the Geological Disposal of Radioactive Wastes, Technical Report Series Number 413, IAEA, Vienna, 2003.

84 European Commission, Geological Disposal of Radioactive Wastes Produced by Nuclear Power - from Concept to Implementation, European Commission Report EUR 21224, Luxembourg, 2004.

85 Nuclear Energy Agency, Lessons Learnt for Ten Performance Assessment Studies, OECD, Paris, 1997.

86 Sumerling T. An international Comparison of Disposal Concepts and Postclosure Assessments for Nuclear Fuel Waste Disposal, A report prepared by Safety Assessment Management for AECL, Whiteshell Laboratories, Canada, AECL TR-M-43, April 1996.

87 International Conference on Geological Repositories: A Common Objective, A Variety of Paths (ICGR'07), Berne, Switzerland, October 15-17, 2007. http://www.icgr2007.org/

88 United Nations Scientific Committee on the Effects of Atomic Radiation, UNSCEAR 2000 Report to the General Assembly with scientific annexes:

Annex A: Dose assessment methodologies;

Annex B: Exposures from natural radiation sources;

Annex C: Exposures from man-made sources of radiation;

Annex E: Occupational radiation exposures (158 pages)

http://www.unscear.org/unscear/en/publications/2000_1.html

89 International Atomic Energy Agency. Survey of wet and dry spent fuel storage, IAEA-TECDOC-1100, Vienna, July 1999.

90 International Atomic Energy Agency. Spent fuel performance assessment and research, IAEA-TECDOC-1343, Vienna, March 2003.

91 Yim M-S and Caron F., Life cycle and management of carbon-14 from nuclear power generation, Progress in Nuclear Energy 48 (2006) 2-36.

92 Van Konyenburg R.A., Behaviour of carbon-14 in waste packages for spent fuel in a tuff repository, Waste Management, Vol 14, No.5 pp.363-383, 1994.

93 International Atomic Energy Agency, Remote technology applications in spent fuel management, IAEA-TECDOC-1433, Vienna, March 2005.

94 USNRC, Spent Fuel Project Office, Interim Staff Guidance-22, Potential rod splitting due to exposure to an oxidising atmosphere during short-term cask loading operations in LWR or other uranium oxide based fuel. www.nrc.gov/reading-rm/doc-collections/isg/isg-22r12.pdf

95 International Atomic Energy Agency, Developing Multinational Radioactive Waste Repositories: Infrastructural Framework and Scenarios of Cooperation, IAEA-TECDOC-1413, Vienna, October 2004.

96 International Atomic Energy Agency, Guidance and considerations for the implementation of INFCIRC/225/Rev.4, The Physical Protection of Nuclear Material and Nuclear Facilities, 1AEA-TECDOC-967 (Rev.1), Vienna, May 2000.

97 International Atomic Energy Agency, Handbook on the Physical Protection of Nuclear Material and Nuclear Facilities, 1AEA-TECDOC-1276, Vienna.

98 Parliamentary Office of Science and Technology, Assessing the risk of terrorist attacks on nuclear facilities, POST Report 222, London, July 2004.

99 Nuclear Information and Resource Service, Armor Piercing Missile Perforates High-Level Radioactive Waste Storage/Transport Cask In U.S. Army Aberdeen Proving Grounds Test, http://www.nirs.org

100 United States General Accounting Office (GAO), Spent Nuclear Fuel: Options Exist to Further Enhance Security, Report to the Chairman, Subcommittee on Energy and Air Quality, Committee on Energy and Commerce, U.S. House of Representatives, July 2003. http://www.gao.gov/new.items/d03426.pdf

101 Sandia National Laboratory, Projected Source Terms for Potential Sabotage Events Related to Spent Fuel Shipments, SAND 99-0963, a report prepared at the request of the Department of Energy, Albuquerque, N.M., June 1999.

102 US National Academies Board on Radioactive Waste Management, Safety and Security of Commercial Spent Nuclear Fuel Storage: Public Report, National Academies Press, 2006. http://books.nap.edu/openbook.php?record_id=11263&page=R1

103 US Nuclear Regulatory Commission, Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants, NUREG-1738, Washington, D.C., February 2001.

104 Sustainable Development Commission, The role of nuclear power in a low carbon economy Paper 6: Safety and security An evidence-based report by the Sustainable Development Commission with contributions from Large & Associates and AMEC NNC March 2006.

105 International Atomic Energy Agency, Multilateral Approaches to the Nuclear Fuel Cycle, Expert group Report of the Director General of the IAEA, Vienna, 2005.

106 International Atomic Energy Agency, The Long-term Storage of Radioactive Waste: Safety and Sustainability, A Position Paper of International Experts, International Atomic Energy Agency, Vienna, 2003.