DISPOSAL OF RADIOACTIVE WASTE

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NEA

The Cost of High-Level Waste Disposal in Geological Repositories

An Analysis of Factors Affecting Cost Estimates

NUCLEAR ENERGY AGENCY

OECD
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ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

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- to achieve the highest sustainable economic growth and employment and a rising standard of living in Member countries, while maintaining financial stability, and thus to contribute to the development of the world economy;
- to contribute to sound economic expansion in Member as well as non-member countries in the process of economic development; and
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The original Member countries of the OECD are Austria, Belgium, Canada, Denmark, France, Germany, Greece, Iceland, Ireland, Italy, Luxembourg, the Netherlands, Norway, Portugal, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The following countries became Members subsequently through accession at the dates indicated hereafter: Japan (28th April 1964), Finland (28th January 1969), Australia (7th June 1971) and New Zealand (29th May 1973). The Commission of the European Communities takes part in the work of the OECD (Article 13 of the OECD Convention).

NUCLEAR ENERGY AGENCY

The OECD Nuclear Energy Agency (NEA) was established on 1st February 1958 under the name of the OEEC European Nuclear Energy Agency. It received its present designation on 20th April 1972, when Japan became its first non-European full Member. NEA membership today consists of all European Member countries of OECD as well as Australia, Canada, Japan and the United States. The Commission of the European Communities takes part in the work of the Agency.

The primary objective of NEA is to promote co-operation among the governments of its participating countries in furthering the development of nuclear power as a safe, environmentally acceptable and economic energy source.

This is achieved by:

- *encouraging harmonization of national regulatory policies and practices, with particular reference to the safety of nuclear installations, protection of man against ionising radiation and preservation of the environment, radioactive waste management, and nuclear third party liability and insurance;*
- *assessing the contribution of nuclear power to the overall energy supply by keeping under review the technical and economic aspects of nuclear power growth and forecasting demand and supply for the different phases of the nuclear fuel cycle;*
- *developing exchanges of scientific and technical information particularly through participation in common services;*
- *setting up international research and development programmes and joint undertakings.*

In these and related tasks, NEA works in close collaboration with the International Atomic Energy Agency in Vienna, with which it has concluded a Co-operation Agreement, as well as with other international organisations in the nuclear field.

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LES COUTS DE L'ÉVACUATION DES DÉCHETS HAUTEMENT RADIOACTIFS DANS DES FORMATIONS GÉOLOGIQUES

Analyse des facteurs influant sur les estimations des coûts

FOREWORD

This study presents an international review of cost estimates for the disposal in geological repositories of spent fuel or reprocessing wastes (high-level vitrified waste and long-lived alpha-bearing waste from reprocessing). The objective of the study is to provide better understanding of the origins of wide variation of the cost estimates, and to demonstrate to what extent various political, institutional, technical and economical factors could explain the variation. The report is intended for the general reader with an interest in the topic.

The work has been carried out by an international group of experts under the auspices of the Nuclear Energy Agency's Committee for Technical and Economic Studies on Nuclear Energy Development and the Fuel Cycle (NDC). The report does not necessarily represent the views of Member governments or participating organisations. The report is published on the responsibility of the Secretary-General of the OECD.

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EXECUTIVE SUMMARY

The disposal of spent fuel or reprocessing wastes (*i.e.* high-level vitrified waste and alpha-bearing waste from reprocessing) in geological repositories has been studied for many years in all OECD countries with a nuclear programme. These studies have demonstrated that safe disposal is feasible in many different types of geological media and that both short-term and long-term safety can be evaluated with acceptable confidence.

Although the actual disposal of such waste is not planned to start in any country until the beginning of the next century at the earliest, rather detailed engineering studies and cost estimates have been made. These studies have the purpose of providing adequate support for planning purposes and for establishing a relevant cost for disposal to be factored into the charge to the electricity consumer.

Although the contribution of the disposal cost to the total electricity generation cost is small, the absolute value of the cost for disposal is expected to be substantial. Some development costs will occur appreciably before disposal, but most will appear long after the corresponding electricity generation. In most OECD countries with a nuclear programme, funding schemes have therefore been established to make provisions for these costs. Fairly accurate estimates of the costs are thus required.

In this study, an international review of reported cost estimates for the disposal of spent fuel or reprocessing waste has been made. As the general situation and circumstances differ substantially between the countries, the bases for the cost calculations are very different and consequently the total estimated disposal costs vary widely.

The objective of the study has been to provide a better understanding of the origins of variations in the cost estimates and to discuss to what extent technical, political, social and economic factors could explain the variation.

Disposal concepts

For the management of spent fuel, two alternative approaches are considered in OECD countries: direct disposal of suitably packaged spent fuel as waste, and reprocessing of the spent fuel to recover the useful products it contains (uranium and plutonium), followed by disposal of the remaining waste products. Direct disposal of spent fuel is the main option in Canada, Finland, Spain, Sweden and the United States, while reprocessing is the main option in Belgium, France, Japan, the Netherlands, Switzerland and the United Kingdom. In Germany, both direct disposal and reprocessing are considered.

The wastes to be disposed of in the two approaches are physically quite different but put similar demands on the repository from the point of view of isolation and temperature restrictions. This means that the repository designs considered are quite similar and indeed some repositories will take both types of waste. For this reason disposal concepts for both spent fuel and reprocessing waste are considered in this report. The content of long-lived radioactivity in these wastes is such that deep disposal (at a depth of a few hundred to a thousand metres) is considered necessary. Short-lived wastes that may be disposed of near the surface are not considered in this report.

Several geological media are being considered for disposal, such as crystalline rock, salt, clay and schist. This range of possibilities naturally provides different conditions for the design, construction and operation of a repository. Consequently, a number of different repository designs have evolved in the different countries.

In general, however, there are many similarities in the repository designs. In all cases presented, the disposal system is based upon the multiple-barrier principle, *i.e.*, the use of multiple barriers, such as the waste form, a corrosion-resistant container, sealing systems and the geological medium. For direct disposal of spent fuel, a separate disposal container is always used, while for vitrified high-level waste (which is vitrified in a stainless steel canister) an extra container (overpack) is proposed only in some concepts. In other concepts for the reprocessing approach, the waste form and stainless steel canister are considered to provide an adequate first barrier. Different materials are used for the disposal container, such as iron, stainless steel, titanium, copper and ceramics. For the alpha-bearing waste from reprocessing, no extra disposal container is normally needed.

The reference design of the repository is, in most countries, based on a tunnel-and-drift design, where the waste packages are disposed of in boreholes drilled into the tunnel floors or in the middle of the tunnels and surrounded by a backfill material. In order to limit the temperature rise in the rock due to the decay heat released from the waste, the waste packages are distributed evenly throughout the rock with a certain separation. The detailed designs are dependent on the physical properties of the particular geological medium.

Cost results

Results of cost estimates have been reported internationally on many occasions. The results are presented in different ways, *e.g.*, total costs, discounted total costs, costs per kWh or funding demand. As the assumptions made are rarely described in any detail, such results do not lend themselves to intercomparisons. Depending on local conditions and on the national strategy, the costs could also cover some or all of the other steps in the spent fuel management process, such as interim storage, transportation, and reprocessing, in addition to final disposal. To be able to make a comparison between cost results, it is important to define clearly what is included in the costs.

In this report, twelve sets of undiscounted cost data for the packaging, and disposal of spent fuel or reprocessing wastes have been provided from eleven OECD countries. The data are given in the national currency at the base year of the cost calculation. For reasons of comparison, the costs have been converted in this report to July 1991 US dollars. The costs have been limited to the design, construction, operation, decommissioning and closing of the facilities. The costs for R&D, site screening and site evaluation, which could be a substantial part of the total costs, are not included, as their content varies widely and there is no basis for comparison between the countries. It should be recognised that these costs are more significant in discounted cash flow analyses because they appear early in the cash flow of the project.

Although packaging and disposal of spent fuel or reprocessing waste are new activities that have not yet been performed, the basic components involved in the process are often well understood because of similar applications in other areas of the nuclear fuel cycle or in other industries. The cost estimates are based on fairly detailed design and operation studies. Of course, a certain level of uncertainty exists in the methods proposed and thus in the estimated costs for repository construction and operation. Normally, therefore, rather high contingencies are included in the estimates. Furthermore, it should be noted that the present cost estimates are, by and large, preliminary. Therefore, great caution should be applied in the interpretation and comparison of the cost data.

As the disposal systems presented vary substantially in capacity, from waste corresponding to an electricity production of 430 TWh (1 800 tonnes of uranium) in the case of Finland to 23 000 TWh (96 000 tonnes of uranium equivalent) in the case of the US, the total costs will also be very different. There is also a difference between disposal of spent fuel and disposal of reprocessing waste, the latter generally being lower in cost. The total costs reported vary between US$ 0.8 and 10.0 billion for spent fuel disposal and between US$ 0.5 and 6.3 billion for disposal of reprocessing waste. It should be emphasized that the costs of reprocessing are not included in these cost estimates. The main cause of these variations is the size of the system, *i.e.*, the amount of waste to be taken care of.

If the costs are normalized with regard to total electricity production, the differences between the different estimates decrease. The normalized cost varies between $ 0.43 and $ 1.77 M/TWh for spent fuel disposal and between $ 0.25 and $ 1.65 M/TWh for disposal of reprocessing waste. A large part of this variation is due to the size effect. As a rather large fraction of the disposal costs are fixed, the specific costs will decrease with the size of the system. The usually applied fuel cycle normalization of "per tonne of uranium" is shown to introduce some major distortions if wastes from different reactor types are compared, and should be avoided.

Technical factors affecting costs

From the earlier description, it is quite clear that other factors also affect costs. As the data provided are not detailed enough to analyse them quantitatively, they are discussed qualitatively in the report.

The most important technical factors, after the size of the system and the choice between direct disposal and reprocessing, are the time schedule of the disposal project, the choice of geological medium and the barrier system chosen.

In most concepts there is a limitation in the maximum allowed thermal impact on the repository and surrounding structures. As the heat output from the fuel and high-level vitrified waste decreases with time as a result of radioactive decay, the waste can be disposed of in a more compact way following longer interim storage time. The time schedule can therefore have a significant impact on the disposal costs.

The different geological media considered have different geotechnical properties that will affect the construction and operation of a repository. In hard rock, for example, big tunnels can normally be kept open for a long time without installation of high-strength liners while in clay, the tunnels must be lined. Differences like these affect the design, construction and costs.

The most important difference between the various concepts concerning the barrier system is whether a separate disposal container is used or not, the material of the container and the type of packaging process. If a separate container is not used, the costs for packaging, which are substantial, do not arise.

Non-technical factors affecting costs

As the disposal of spent fuel or reprocessing waste is expected to be a highly controversial political issue in most countries, the social and political issues will inevitably affect the costs. They will affect, for example, the siting and licensing process (by political delays, demands for further investigations and procedural complications, etc.) as well as the overall waste management policy.

The effect of these social and political factors cannot easily be included as a straightforward cost factor in the cost calculations, but could be accounted for as an extra risk factor. Other social and political factors such as taxes, the cost of land, compensation and mitigation of impacts on the local population and the environment could, however, easily be included.

In the comparison of cost estimates, one complicating factor is the economic and financing considerations that are included in the costs as presented. This is particularly true for the funding estimates, where interest and discounting factors are included. As the time span over which the costs for disposal will occur is very long, these factors strongly distort the comparison. In this report, in order to avoid this complication, only undiscounted costs in price value of July 1991 US dollars are used. This means that the specific costs per TWh reported here are greater than the ones used to accumulate funds to cover the cost of disposal.

Conclusions

Cost estimates for the disposal of spent fuel or reprocessing waste have been made in many countries. In this report, twelve different estimates from eleven countries are reported and compared.

The estimates are based on design studies. As no disposal facility will be in operation until the beginning of the next century at the earliest, the costs must be regarded as preliminary and be treated cautiously, as a rather large uncertainty exists. In the cost estimates, a high contingency factor is normally applied. Although no packaging or disposal facilities yet exist, it should be recognised that many of the cost components are based on well-established experience in other nuclear and non-nuclear fields.

It has been determined that comparison of the cost estimates is very difficult and certainly should not be done without taking due account of the different bases on which the cost estimates have been prepared. The comparison is more meaningful after due normalization. In fact, considering the differences in system designs, there is surprisingly good agreement between the estimates when they are normalized with regard to total electricity production and the remaining differences can be explained at least qualitatively. This indicates that the disposal costs are reasonably well understood in the OECD countries.

The cost of disposal of spent fuel or waste from reprocessing is only a small fraction of the total electricity generation cost. The uncertainties in these costs indicated by the variation in the cost estimates presented here will therefore have only a marginal effect on the cost of electricity production from nuclear power.

Chapter 1

INTRODUCTION

Studies of the disposal of spent fuel or of reprocessing wastes* have been performed in all countries with a nuclear power programme. The development and demonstration of the technology for disposal and licensing is progressing well but operating disposal facilities will not be available in OECD countries until the beginning of the next century at the earliest. Extending the time between fuel discharge from a reactor and disposal provides the technical advantage of reducing the heat and radiation emissions from the waste due to radionuclide decay, offering the possibility of simplifying the design and reducing the size of a disposal facility.

Spent fuel or reprocessing wastes are highly radioactive, long-lived or toxic, and could represent a significant hazard to man and his environment, if not managed properly. It is essential that they be isolated from the biosphere for very long periods of time. The generally accepted method for isolation is disposal at depth in geological formations that provide a stable environment for a very long time.

The safety of disposal of such wastes has been described and discussed in numerous reports, including other NEA publications. Although no disposal facility has yet been constructed, no major technological problems are expected, as the technology to be used is based almost entirely on experience from other existing nuclear facilities, *e.g.*, reprocessing plants and disposal facilities for short-lived wastes, and from other non-nuclear areas.

The cost of disposal is expected to be substantial and will appear long after the corresponding electricity generation. In most OECD countries with nuclear power programmes, funding schemes have, therefore, been established to cover the future costs and a considerable amount of money has already been set aside.

As a basis for the funding schemes, estimates of the costs for disposal have been produced in OECD countries. As the conditions vary substantially between the countries, the results of these estimates have also shown a great variation. Therefore, it could be suggested that the costs of final disposal are not well founded. However, closer examination reveals good reasons for the differences, such as variations in the technical/engineering aspects and national disposal strategies, and non-technical issues such as schedule and financing assumptions.

Although the total costs are substantial, the contributions of the disposal costs to the total fuel cycle costs are small. A previous NEA study on nuclear fuel cycle costs [1] suggests that disposal costs in levelised cost calculations represent only a small percentage of the total fuel cycle costs. Therefore, even if there are significant fluctuations in the disposal cost estimates, they have only a small impact on the total cost of electricity generated by nuclear power stations.

Accurate estimates of the cost of disposal are, however, important for calculating the contribution to the funding scheme required to fund fully the cost of future disposal. This contribution is collected in the price charged for the sale of electricity. Too low a contribution will impose an economic burden on future generations, while too large a contribution will impose an undue burden on present electricity consumers. General equity considerations thus require that the estimated costs be as close as possible to the eventual costs.

The cost estimates are based on considerable technical knowledge from nuclear and non-nuclear activities and established geotechnical engineering experience. Therefore, although the cost estimates include a measure of uncertainty, major causes of the variations of the estimates relate to technical issues such as size of the nuclear power programme, waste form, disposal medium, selection of disposal system components, safety regulations, socio-political factors and economic assumptions. However, this has not been clearly and systematically established in previous studies. The NEA's economic studies [1, 2], where disposal costs entered into the total costs considered, did not focus on the causes of the variation.

* In the remainder of this report, the term "reprocessing waste" is used as an abbreviation for "high-level vitrified waste and alpha-bearing waste from reprocessing activities".

Scope and goal

This study provides an international review of cost estimates for disposal of spent fuel or reprocessing waste in geological repositories. The principle objective of this study is to provide a better understanding of the origins of variations in the cost estimates, and to discuss to what extent various political, institutional, technical and economic factors could explain the variations. The main effort has been concentrated on identifying the factors that may affect the cost estimates for the packaging and disposal of waste. Other steps in the spent fuel management system will be discussed but not analysed in detail, as they are strongly dependent on the strategy adopted in each country.

In this study, the costs will be considered both for spent fuel disposal and for disposal of reprocessing wastes. It must, however, be emphasized that these costs should not be used for comparison between the direct disposal and the reprocessing routes, as not all the costs involved will be considered here. For example, in the case of the reprocessing route, neither the cost of reprocessing nor the value of the fissile material recovered is considered. The safety of the disposal will not be discussed in this report but has been covered in other publications [3, 4, 5, 6, 7].

Structure of this report

This report, intended for the general reader with an interest in this topic, discusses various points that should be considered when one examines disposal cost estimates. Chapters 2 and 3 briefly review the spent fuel management systems and disposal systems. Chapter 4 explains the cost components of the disposal. Chapter 5 provides the cost calculation procedures, and the results of cost calculations presented to the Expert Group are given in Chapter 6, which also includes a discussion on normalization methods of the estimates. These estimates form the main reference for the subsequent discussion. Chapter 7 describes the technical factors that affect cost calculations and Chapter 8 describes the non-technical factors. Chapter 9 provides the conclusions of the study. Specific details about cost calculations for current waste disposal strategies being studied in some OECD countries are given in Annex 1 at the end of the report. Annex 2 provides a list of abbreviations and glossary of terms used in the report.

Participants

This study has been undertaken under the auspices of the NEA's Committee for Technical and Economic Studies on Nuclear Energy Development and the Fuel Cycle (NDC). The preparation of the study has been overseen by an Expert Group whose membership is listed in Annex 3. Twelve OECD countries and two international organisations have participated in the study.

Decommissioning costs

A similar study has been performed for decommissioning cost estimates. The objective of that study was to make clear the causes of the variation in the cost estimates for decommissioning nuclear power plants. The study report was issued September 1991 [8].

Chapter 2

SPENT FUEL MANAGEMENT SYSTEM

2.1. General

The management of spent fuel starts with the discharge of the fuel from the reactor core to fuel storage in the nuclear power plant and ends with the disposal of the fuel or its waste residues. Two general approaches to spent fuel management are being considered:

- **the reprocessing approach,** in which the spent fuel from the reactor is reprocessed, to separate plutonium and uranium, which can be reused as nuclear fuel, from other radioactive elements produced in the fission process in the reactor core;
- **the direct disposal approach,** in which the spent fuel is not reprocessed but is disposed of as a waste product following appropriate treatment.

The selection of an approach to spent fuel management by a country or utility is based on the consideration of a number of factors. These may include national nuclear strategy, regulations, costs and social effects. As each of these factors may have different meanings and implications in each country, the spent fuel management strategies may vary both in technical detail and in schedule.

Irrespective of the approach chosen, a number of steps and actions must be taken in order to manage the spent fuel safely. In Figure 2.1, the different stages of spent fuel management are illustrated schematically. The figure also provides an indication of the quantities of the material involved in the different stages for each one-year operation of a 1 000 MWe pressurized water reactor (PWR). As the mass balance is affected by various technical factors, the quantities indicated in Figure 2.1 are for illustration only.

This economic study focuses on the costs for packaging and disposal of spent fuel in the direct disposal approach, and of reprocessing wastes in the reprocessing approach. The costs for reprocessing, interim storage and transport are not considered in this study. If a cost comparison between the two spent fuel management approaches is the objective, the costs for all stages of spent fuel management, as well as the value of the recycled material, must be included. Such a comparison has been done by others [1, 9, 10].

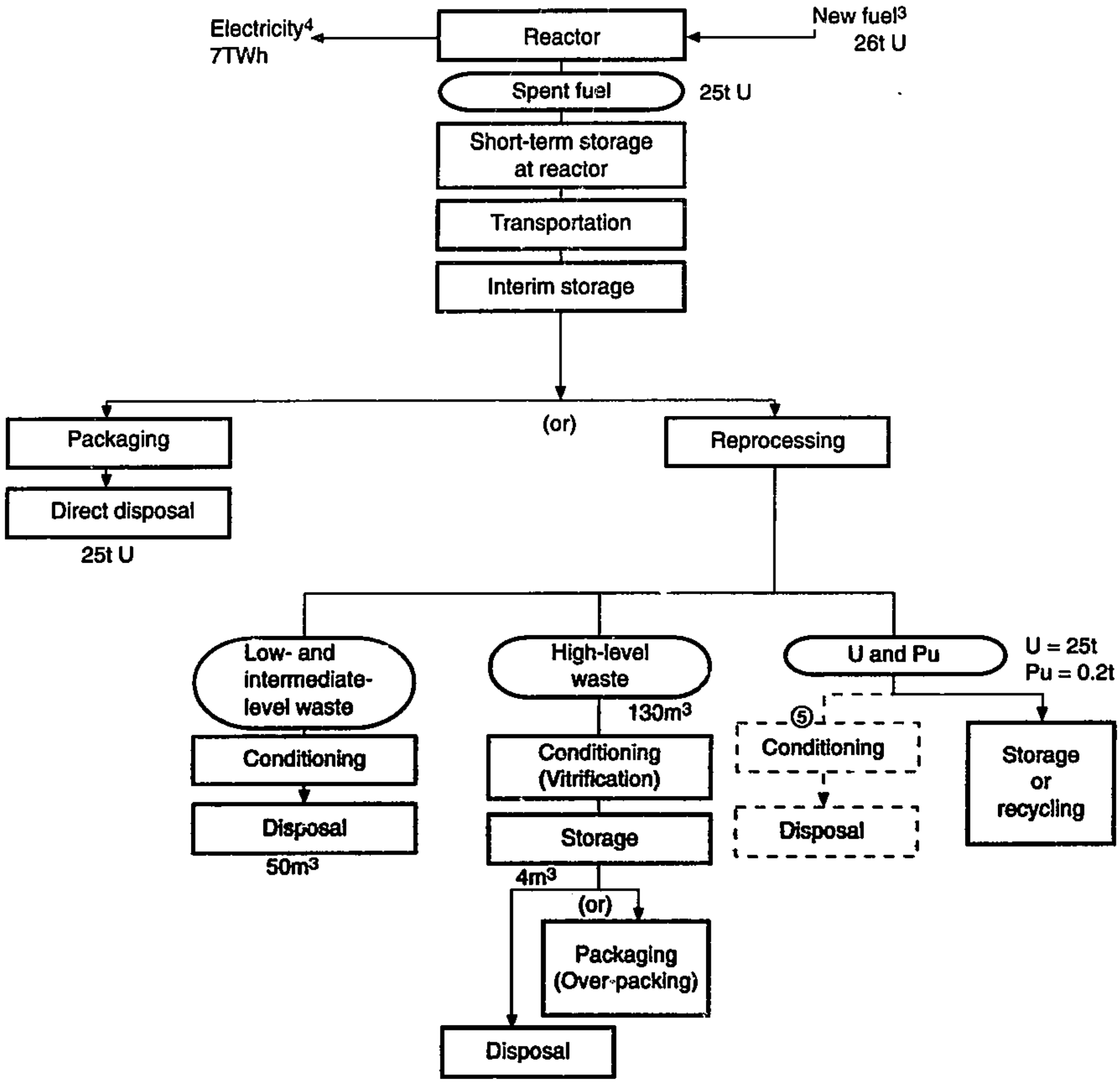
In the remainder of this chapter, an overview is given of the main components of the spent fuel management system, and the characteristics of the spent fuel and the wastes to be considered are described. A more detailed explanation of the final disposal stage, including packaging as needed, is given in Chapter 3.

2.2. Spent nuclear fuel

The fuel for a light water reactor (LWR) consists of cylindrical pellets of sintered uranium dioxide enclosed in rods of zircaloy. The rods are bound together in fuel assemblies that are handled as units. A fuel assembly for a 1 000 MWe PWR typically contains about 250 fuel rods and has a length of about 4 m. Figure 2.2 shows a typical PWR fuel assembly. The content of uranium is about 500 kg per assembly. The uranium is initially enriched to 2-4 per cent uranium-235.

Other reactor types have other fuel designs. Some data of typical fuel assemblies are given in Table 2.1. A boiling water reactor (BWR) fuel assembly is very similar to that of a PWR but has fewer rods and only about 200 kg of uranium. A heavy-water reactor (HWR) fuel bundle is about 50 cm long and 10 cm in diameter, consisting of 19 to 37 individual cylindrical pins of natural uranium dioxide, and containing about 19 kg of uranium per assembly. In advanced gas-cooled reactors (AGR), uranium dioxide fuel encased in stainless steel rods is used. The assemblies are about 1 m long and contain 36 rods in a graphite cylindrical sleeve, totalling about 42 kg of uranium enriched to 2.5 per cent. In Magnox gas-cooled reactors, uranium metal encased in a

Figure 2.1[1] **Stages of spent fuel management and quantities[2] of the materials involved in the different stages**



1) This figure is prepared on the basis of an NEA report; *The Economics of the Nuclear Fuel Cycle*, Paris, 1985.
2) Quantities for each one-year operation of a 1000 MWe PWR.
3) 3.1% enrichment, for 33000 MWd / t U.
4) With a load factor of 80%.
5) Recovered U and Pu may be disposed of after recycling a number of times.

Figure 2.2 **A new fuel assembly of a PWR**

Photo: Électricité de France

magnesium alloy is used as fuel. The fuel elements are 1 m long and contain about 12 kg of natural uranium. Other less common types of spent fuel come from research reactors, fast breeder reactors, etc.

When the fuel assembly is discharged from the reactor, it is still handled as a unit and its appearance is generally similar to fresh fuel. The fuel, however, is highly radioactive and emits radiation and heat, as it contains uranium, plutonium, fission products and other actinides, such as neptunium, americium and curium. A typical

Table 2.1. **Characteristics of typical fuel assembly**[a]

	PWR[b]	BWR[c]	CANDU	Magnox	AGR
Approximate fuel assembly dimensions (cm)					
- Length	435	447	50	100	100
- Cross section					
Side (square)	23	14			
Diameter (cylinder)			10	10	24
Fuel weight (kgU/assembly)	531	184	19	12	42
Average fuel enrichment (w/o)	2.4/3.5	3.1	0.7	0.7	2.5
Average burnup (MWd/kgU)	42	36	8	5.5	18

a) Prepared on the basis of "Guidebook on Spent Fuel Storage, Second edition", Technical Report Series No. 240, IAEA, Vienna, 1991.
b) Rod array: 18 x 18. Number of rods: 300.
c) Rod array: 9 x 9. Number of rods: 76.

LWR spent fuel (3.1 per cent enrichment; 33 000 MWd/tU* burnup) consists of about 96 per cent uranium, 1 per cent plutonium and 3 per cent fission products and other actinides [1]. These materials are contained in the uranium dioxide fuel matrix. For a 1 000 MWe PWR, typically 25 tU** of spent fuel is generated annually.

The radioactivity of the spent fuel decreases with time. At first, the decrease is very rapid as the short-lived radionuclides decay. As time passes, the radioactivity decreases more slowly, as it is controlled by the decay of the more long-lived radionuclides. The decrease in radioactivity results in a corresponding decrease in heat generation. In Figure 2.3, the evolution of heat generation with time after discharge from the reactor is shown for typical PWR spent fuel. One year after discharge, the radioactivity content is about 73 000 TBq/tU, and the heat generation is about 8.1 kW/tU. In Figure 2.3 also, the evolution of heat generation with time for typical high-level vitrified waste is shown.

Other fuel types will have other burnups and, consequently, other absolute levels of radioactivity and heat generation. Typical examples of heat generation at one year after discharge are: 3.2 kW/tU for CANDU fuel (approx. 8 000 MWd/tU), 1.0 kW/tU for Magnox fuel (approx. 5 500 MWd/tU) and 4.0 kW/tU for AGR fuel (approx. 18 000 MWd/tU). The time behaviour of these quantities will, however, be pro rata to the PWR fuel.

2.3. Outline of spent fuel management

2.3.1. General

As was shown in Figure 2.1, a spent fuel management system typically includes some or all of the following steps:

- storage at reactor;
- transportation;
- interim storage;
- reprocessing;
- packaging/conditioning;
- final disposal.

In the following paragraphs, each of these steps is briefly described.

2.3.2. Storage in spent fuel cooling pond at the reactor site

After being discharged from the reactor core, the spent fuel is placed in a pond or in dry storage where it is cooled and its radiation field is contained by shielding. While in storage at the plant, the short-lived fission products decay rapidly and the heat output decreases correspondingly. With LWR fuel, for example, the heat from a fuel assembly (0.46 tU, 33 000 MWd/tU) is 17 kW after one month, 4 kW after one year and 0.8 kW after five years from the time of discharge from the reactor.

The length of the cooling period at the reactor may vary from less than a year to a few decades, depending on the national nuclear policy, the availability of a reactor or an interim storage capacity, the reprocessing capacity and/or the disposal facility. If a long on-site storage period is planned, the fuel will in some cases be transferred from a reactor storage pond to a dry storage facility or an auxiliary wet storage facility. Dry and wet storage facilities are already in operation. Details of at-reactor storage are reported in, for example, IAEA reports [11, 12].

2.3.3. Transportation

After the initial period of spent fuel storage at the reactor site, transportation is an essential part of spent fuel management, irrespective of the approach chosen. The transportation of spent fuel is a well-established practice that has been performed on a routine basis for more than 20 years. Transport is by truck, rail, or ship. The transportation standards are covered by the IAEA Regulations for the Safe Transport of Radioactive Materials [13] and controlled by specific regulations issued by individual governments. These regulations require,

* In this report, burnup is consistently given in MWd/tU, although MWd/tHM (tonne of heavy metal) is more precise.

** In this report, the amount of nuclear fuel is consistently given in tU (tonne of uranium), although tHM (tonne of heavy metal) is more precise.

Figure 2.3 **Evolution of decay heat from 1 tU of spent PWR fuel (irradiated to 33 000 MWd/tU) and corresponding high-level vitrified waste**

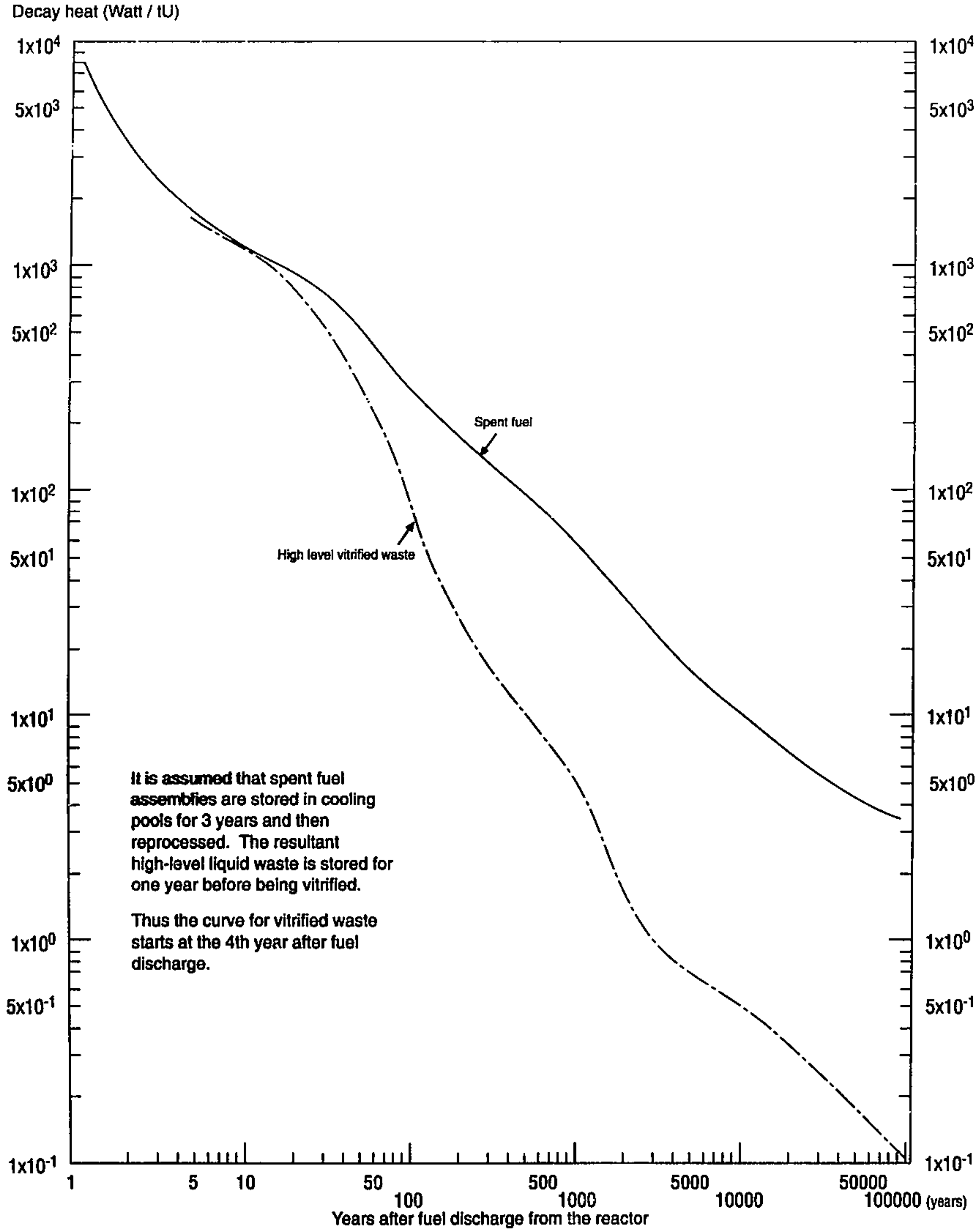
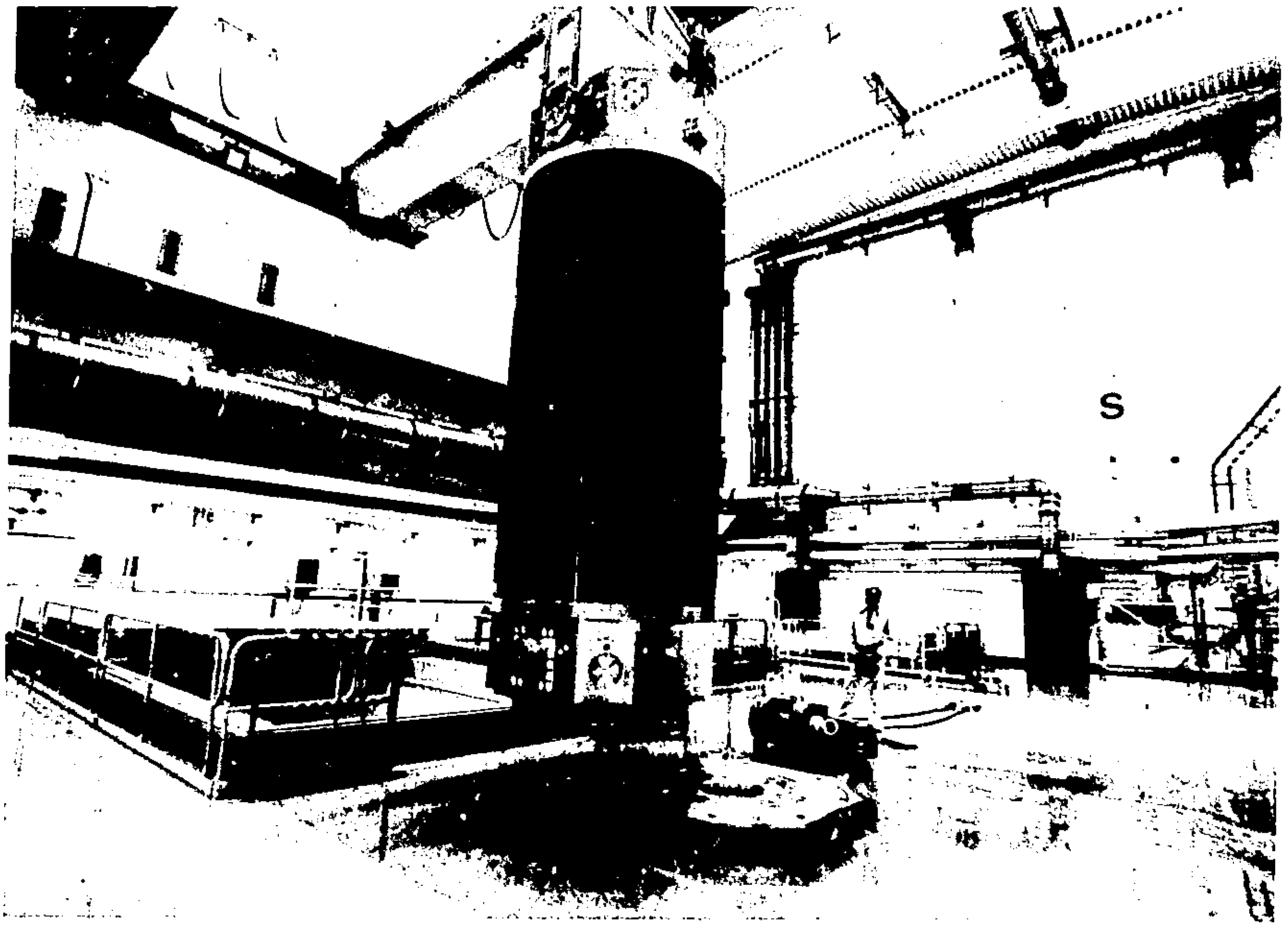


Figure 2.4 **Transport cask for spent fuel (Sweden)**

Photo credit: Mr. B. Emmark



among other things, that a prototype of each transport cask undergo specific tests that simulate severe accident conditions as part of the licensing process.

A transport cask for spent fuel is a massive box or cylinder weighing 50-120 tonnes that can hold 1-8 tonnes of fuel. The thick cask walls made of steel, together with shielding made of steel, depleted uranium and/or a material containing hydrogen, such as polyethylene or paraffin wax, provide ample radiation shielding for gamma and neutron radiation. The casks are also designed to dissipate the heat generated in the fuel. The need for radiation shielding and heat dissipation decreases with time and the design of the cask and safety case will relate specifically to a heat load that is a function of fuel mass and cooling time. Figure 2.4 shows a typical transport cask.

The transport casks for high-level vitrified waste and some of the alpha-bearing wastes from reprocessing will be of similar design to those for spent fuel, but specially designed for the radiation, heating and geometric characteristics of the waste package to be transported.

2.3.4. Interim storage of spent fuel

In some strategies of spent fuel management, spent fuel will be transferred from the cooling ponds at the reactor site to interim storage facilities away from the reactor site and stored there for some time before reprocessing (in the reprocessing approach) or conditioning prior to disposal (in the direct disposal approach). The necessity for the interim storage and the length of the storage period is determined by the capacity of the storage facilities at the reactor and the availability of the reprocessing capacity or of the disposal facility. In some national strategies, extended storage is often used to allow radioactive decay to reduce the heat generation of the spent fuel before disposal, thereby changing the specifications for the disposal system.

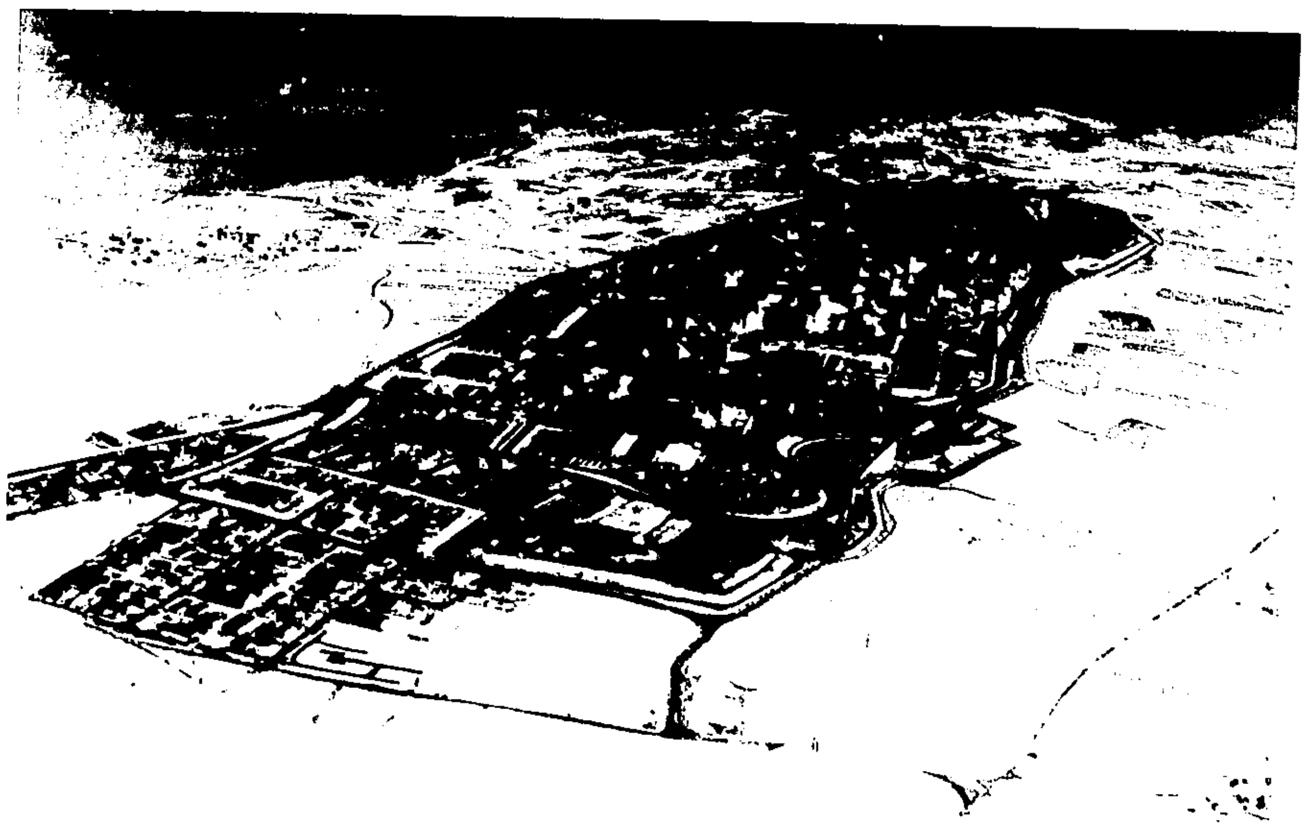
In cases where the cooling pond at the reactor site has sufficient capacity or where the pond or dry storage capacity could be reasonably expanded, interim storage could take place at the reactor. In this case, the spent fuel could be kept in interim storage at the reactor site for several decades.

The location of an interim storage facility will be dependent on national circumstances. It is often co-located with a reactor and it could serve one reactor or all the reactors in the country. Alternatively, it could be located at the reprocessing or disposal site or at a separate location. In some countries, a considerable amount of fuel will accumulate and will be stored for a relatively long period. This may favour the development of large-scale central facilities dedicated to the storage of spent fuel and seeking to take advantages of economies of scale, although requiring additional waste transportation.

Various approaches have been developed for interim storage [14, 15]. In some approaches, the fuel assemblies are stored in ponds where they are cooled by water. In other approaches, the fuel assemblies can be safely held in dry storage (*i.e.*, dry pits, dry casks, etc.) where cooling is accomplished using either air or inert gases with natural or forced circulation. The fuel assemblies are normally stored intact. In some cases, the assemblies are disassembled (rod consolidation) to achieve a closer packing and hence a volume reduction. In many cases, the waste is sealed in specially designed storage containers.

2.3.5. Reprocessing

The technology of spent fuel reprocessing is well-established and used on a commercial scale in F ace and the UK. The purpose of reprocessing is to separate uranium and plutonium from other actinides and ıssion products contained in the spent fuel. The fuel is dissolved in nitric acid, and uranium and plutonium are separated in a chemical process (*e.g.*, the PUREX process). The uranium and plutonium recovered are recycled for possible subsequent use as nuclear fuel. The rest (other actinides, fission products and impurities) become a highly radioactive solution (high-level liquid waste) and are stored for further conditioning. Figure 2.5 shows an aerial view of the La Hague reprocessing plant in France.



During operations at the reprocessing plant, several separate categories of radioactive waste are produced.

The **high-level liquid waste** contains more than 99 per cent of the non-gaseous fission products, together with traces of plutonium and other actinides. The waste may be concentrated by evaporation and stored in stainless steel tanks, which are water-cooled, double-walled, and situated in shielded facilities.

For interim storage and final disposal, the waste is converted to a stable solid form. The most common solidification method is vitrification with borosilicate glass in a stainless steel canister. Other processes involving glass or ceramics are considered. Vitrification processes and the characteristics of the vitrified materials have been studied intensively and the methods have been adopted for industrial-scale operation in many countries (Belgium, France, Germany, Japan, the UK). Vitrification provides a low-volume solid waste form. Vitrified waste is chemically durable and has suitable physical and thermal properties for long-term storage and disposal. An artist's impression of vitrified high-level waste is shown in Figure 2.6.

The amount of vitrified waste from a 1 000 MWe PWR will typically be about 4 m^3/year. The dimensions, chemical characteristics and radioactivity of the waste are affected by the methods and specifications of the vitrification process. Table 2.2 provides some information on typical vitrified waste.

The rate of radioactive decay and the decrease in heat generation of the vitrified waste is at first comparable to that of the spent fuel, as the radioactivity is dominated in both cases by the fission products. After some time, the plutonium nuclides and their daughter products come to dominate the decay of the spent fuel, and the decay of vitrified waste then becomes more rapid, as shown in Figure 2.3.

Intermediate-level liquid wastes are usually contaminated with alpha-emitting radionuclides. The wastes can be processed to concentrate their radioactive content, which can then be added to the high-level waste stream, or alternatively immobilised into a solid matrix such as concrete, bitumen, or resin. **Low-level liquid wastes** contain very little radioactivity and are disposed of after appropriate treatment or discharged under carefully controlled conditions.

Solid wastes include the cladding removed from the spent fuel, filters, resins and other materials used during the reprocessing operation, and contaminated plant and equipment. The wastes are radioactive to various

Figure 2.6 **Artist's impression of vitrified HLW in a stainless steel canister**
Nuclear Electric diagram from BNFL photo

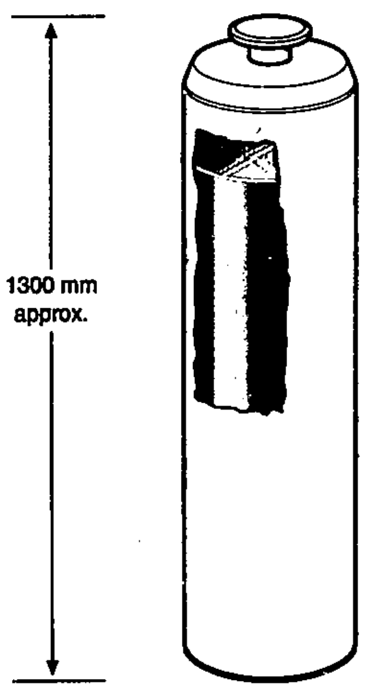
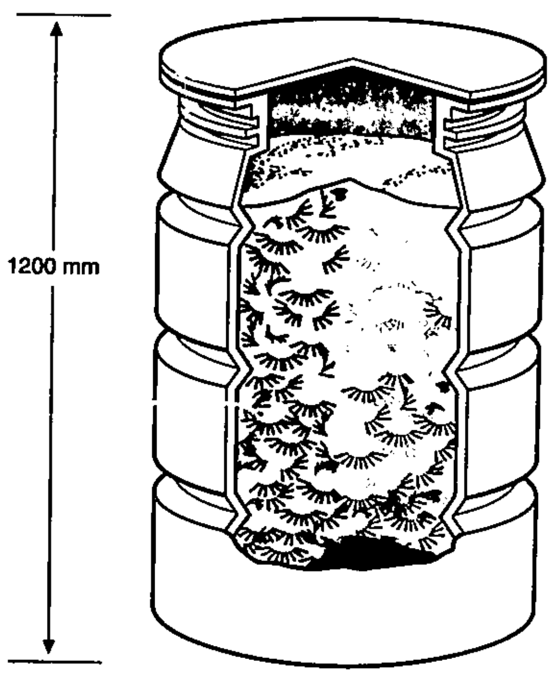


Table 2.2. **Characteristics of typical high-level vitrified waste**[a]

Approximate dimensions (cm) (cylinder)	
- Length	134
- Diameter	43
Capacity (l)	
- Nominal	170
- Glass	150
Weight (kg)	
- Total (canister and waste)	490
- Canister	80
Solid matrix	Borosilicate glass
Material of canister	Stainless steel
Radioactivity at the time of vitrification[b]	alpha: 140 TBq/canister beta-gamma: 28 000 TBq/canister
Heat generation rate at the time of vitrification	3 000 W/canister

a) COGEMA specification.
b) It is assumed that spent fuel is stored in cooling pool for 3 years before reprocessing and that the resultant high-level liquid waste is vitrified after 1 year of storage.

Figure 2.7 **Artist's impression of reprocessing ILW (finned Magnox fuel cladding) solidified with cement in a stainless steel package.** Nuclear Electric diagram from BNFL photos



degrees and most of them are contaminated with alpha emitters. After possible volume reduction by incineration, compaction or shredding, the wastes are immobilised into a solid matrix such as concrete or metal for disposal. Figure 2.7 shows an artist's impression of such waste.

Gaseous wastes are produced during the chopping up and dissolution of the spent fuel. After removing radioactive particulate materials by filtering and then removing some gaseous wastes by chemical processes, the remaining gases are discharged under carefully controlled conditions to the atmosphere.

Some of the non-high-level waste streams described in the preceding paragraphs contain long-lived radionuclides in quantities that make geological disposal appropriate. The dividing line between wastes with contamination levels suitable for other disposal routes, and those for which geological disposal may be appropriate, depends on their actinide content, their conditioning, the characteristics of the disposal system and national regulations. Because of their relatively low level of radioactivity and low rate of heat generation, these wastes may be handled in a simpler way than high-level wastes and spent fuel. Typically, 50 m^3/year of conditioned alpha-bearing waste is generated from the reprocessing of fuel from a 1 000 MWe PWR.

2.3.6. Interim storage of waste from reprocessing

In most countries that have taken the reprocessing approach, interim storage is also considered for reprocessing wastes during the period between conditioning and final disposal, (*e.g.*, several decades). This storage provides flexibility for the disposal schedule as well as the advantage of a lower heat generation rate in the vitrified waste at the time of final disposal, due to radioactive decay during the storage period. The rate of heat generation decreases by a factor of 50 or more between the first and the hundredth year after reprocessing. The storage facility is, in some countries, assumed to be located at the reprocessing plant site or final disposal site.

2.3.7. Packaging and final disposal

Although no countries have experience with commercial-scale disposal of spent fuel or reprocessing wastes, intensive research and development programmes for disposal have been pursued in almost every country with a nuclear programme.

The only disposal method considered at present is geological disposal, where appropriately packaged wastes will be disposed of in repositories which will be constructed between several hundred and one thousand metres underground. The repository for disposal of radioactive waste must provide a high isolation capability and be adequately stable. The repository design has to be optimised for each site, bearing in mind the type of waste, the type of host rock, site-specific conditions and so on. A more detailed description is given of the disposal methods considered at present in Chapter 3.

The safety of final geological disposal is generally accomplished by the use of multiple barriers, *e.g.*, the waste form, a corrosion-resistant container, a sealing system and the geological media in which the disposal facility is built. In the design of a disposal system, the balance amongst different barriers has to be considered and safety can only be judged from analyses of the entire system. This means that in some cases a thick-walled, long-lived container is used, thereby relieving some of the demands on other barriers, while in other cases no corrosion-resistant container is taken into account in the safety analyses. The latter could, for example, be the case for vitrified waste and some of the alpha-bearing waste, where the container used in waste conditioning is also the disposal container.

For spent fuel disposal, a corrosion-resistant container (more exactly, canister) is normally considered in order to ensure safe handling during the emplacement of the waste and a longer-term containment of radionuclides in the repository in order to limit and delay for a significant period any release of radionuclides from the waste.

The canister may also be constructed so as to provide adequate radiation shielding during handling for manually controlled transport and emplacement. Alternatively, a thinner canister may be used to reduce the amount of non-radioactive material disposed of with the waste but it will require remote handling or an additional overpack or cask for handling purposes.

For the disposal of alpha-bearing waste, there are generally no requirements for additional packaging. The primary packaging is often considered to be adequate for disposal. During handling, however, an extra overpack or cask may, in many cases, be required for shielding purposes.

After waste packages have been emplaced in the repository, the residual space in emplacement drill holes and in excavated tunnels will be filled by backfilling materials, such as spoil from excavations, bentonite, and cement. In many disposal strategies, excavation and preparation of additional disposal tunnels, backfilling and sealing are planned to be carried out concurrently with waste emplacement. When the emplacement of waste and a period of monitoring, if required, have been completed, the access tunnels and shafts will be backfilled and the surface facilities will be decommissioned. The site may be released for unrestricted use. In some countries, however, it is foreseen that long-term institutional control of the area will be necessary.

Chapter 3

GEOLOGICAL DISPOSAL CONCEPTS

3.1. Introduction

Within geological disposal, there are many alternatives. The choice of multiple barriers is important, as described in Chapter 2. Furthermore, the details, and subsequently the costs, are affected by the geological media, national regulations, site conditions, location of packaging facilities, and other factors. This chapter provides a detailed description of the packaging and disposal methods considered in OECD countries.

3.2. Packaging facility concepts

3.2.1. General overview

The packaging of spent fuel and reprocessing wastes involves sealing them in engineered containers* that are designed to have a significant period of structural integrity in the conditions expected at the disposal site. The containers for packaging either spent fuel or reprocessing waste will be designed to satisfy the specific requirements of the national disposal strategy, the regulations in force in that nation and the disposal site conditions. The container structural integrity contributes to a disposal strategy in two ways. First, the container will be an absolute barrier to the release of radionuclides from the waste to the natural environment for the time it takes for the container to corrode through. Further, the container will be structurally sound for an additional period of time during which it can facilitate waste retrieval if this is required.

Of the container concepts considered, two different categories can be distinguished. One category refers to containers with an expected service life of a few hundred years to a few thousand years. Materials discussed for these containers are carbon steel, stainless steel and titanium. The other category refers to containers with expected service life of hundreds of thousands of years. These containers are generally made of copper or ceramics.

In addition, a distinction can be made between a solid container, where all voids in the loaded container are filled with some supporting material, and a self-supporting container, whose walls are strong enough to take up the rock and groundwater pressure without any detrimental deformation.

A third categorisation distinguishes between containers that will provide adequate shielding for handling, and those for which a transfer cask or an overpack will be needed during handling and disposal.

The packaging operation will begin with the receipt of intact spent fuel assemblies from a storage facility or reprocessing waste from a reprocessing facility and end with the shipment of disposal containers to the disposal facility.

3.2.2. Spent fuel packaging

Spent fuel will be shipped from the nuclear generating stations or interim storage facilities to the packaging plant in licensed shipping casks whose capacity will depend on the method of transport (road, rail or ship) and the age of the fuel. The geometry of the spent fuel assemblies has been described in Chapter 2.

At the packaging facility, the shipping casks of spent fuel will be received and unloaded. After unloading, the casks will be resealed, decontaminated and returned to the shipper. The fuel may be packaged as received, or

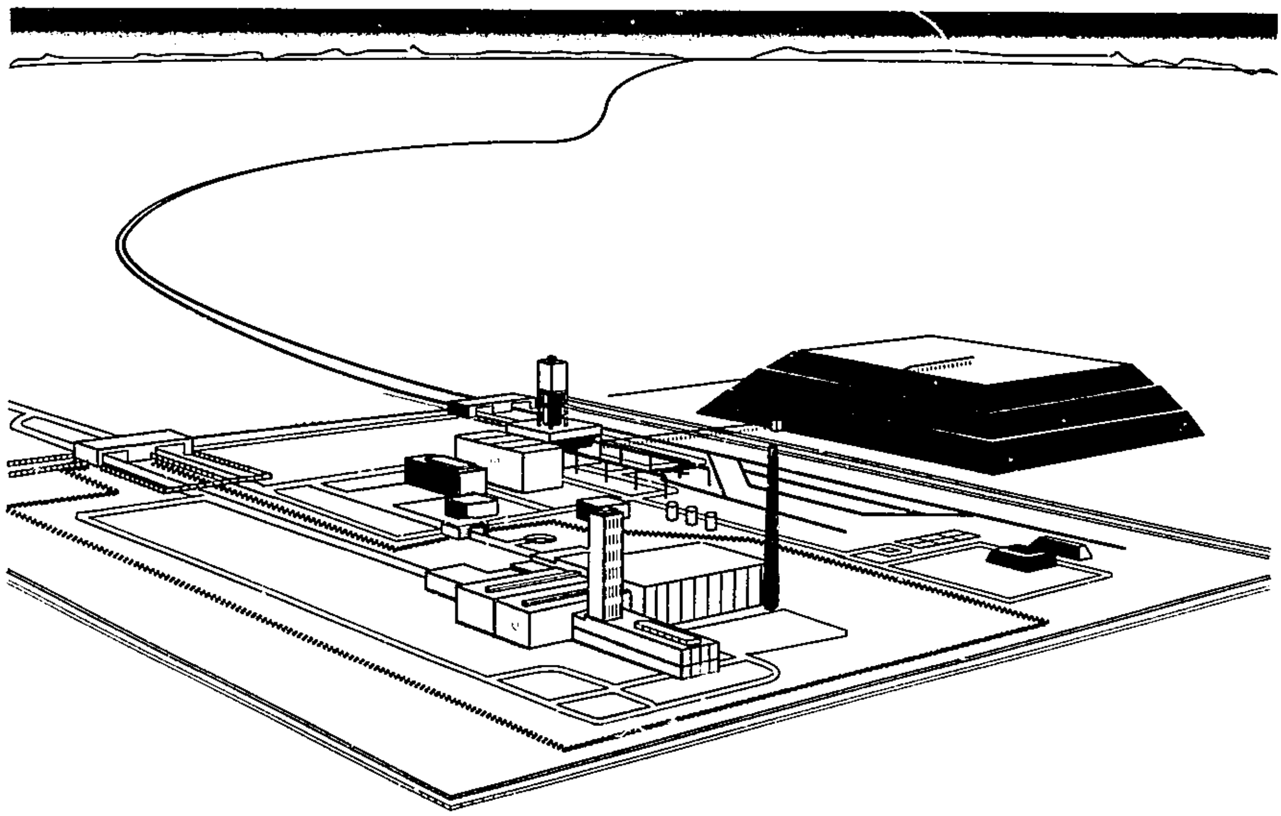
* Hereafter, the word "container" is a generic word for disposal purposes, and includes "canister" and "overpack". A canister is a closed or sealed container for spent fuel or reprocessing waste. An overpack is a secondary external containment and/or shielding for radioactive waste packaged in a canister (see Glossary).

may be disassembled and packaged in a more compact form. In either case, shielded hot cells will be required to allow these operations to be conducted safely and to control the spread of radioactive contamination, particularly if the fuel assemblies are disassembled prior to packaging. Temporary storage facilities are likely to be included to smooth the material flows through the receiving and packaging operations. These could accommodate both the fuel assemblies and the packaged waste.

The packaging process will involve the fabrication, loading, sealing and inspection of disposal containers. The container geometry and design will be a function of the fuel and repository design. The facility will provide the means for identifying the authenticity of the spent fuel for safeguards* purposes, transferring the fuel assemblies or their components to the containers, sealing, inspecting and decontaminating the containers, and storing them for transfer to the repository. If the process chosen includes the disassembly or chopping of fuel assemblies to improve the flexibility of packaging, there will also be fuel assembly hardware and fuel scrap from this processing that must be handled as waste. Most of these operations will have to be performed remotely in hot cells.

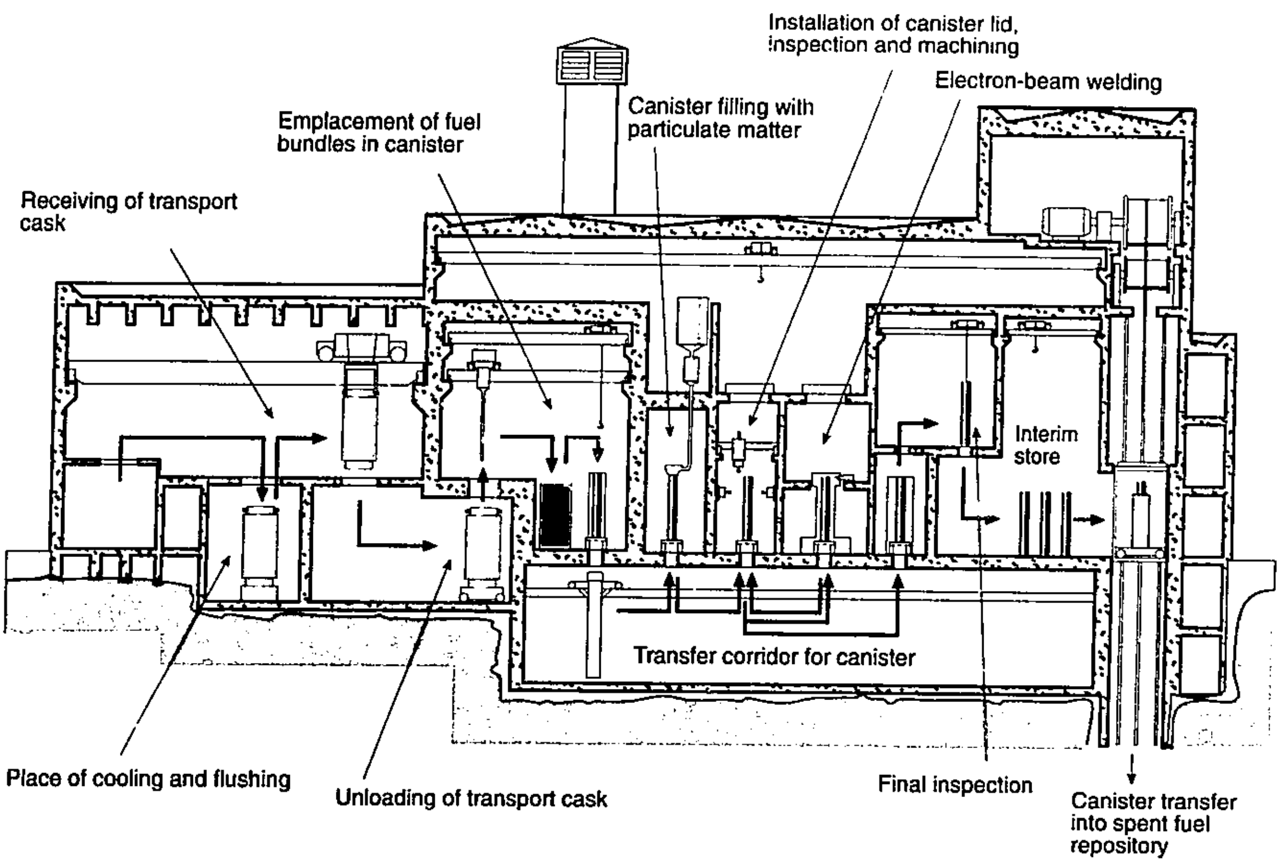
A packaging facility will be a rather large industrial complex. An example of the layout of a packaging facility is shown in Figure 3.1, and in Figure 3.2 a typical flow diagram for a packaging (more specifically "encapsulation") process is shown.

Figure 3.1 **Artist's view of packaging facility and surface facilities**
(Spain, for a salt formation)



* Safeguards are measures to prevent or detect the diversion of nuclear material and to protect against the sabotage of facilities. The safeguards employed by a nation (domestic safeguards), which sometimes cover measures for physical protection, may be different from the IAEA safeguards (international safeguards). The term "safeguards" used here includes both. Please see Glossary (Annex 2).

Figure 3.2 **Typical flow diagram of packaging**
(Finnish encapsulation facility)



Additional supporting site service installations will supply the process utilities, waste treatment, trades, stores and warehousing, administration and management needed to operate the packaging plant.

Packaging of spent fuel has been studied in many countries, *e.g.*, Canada, Finland, Germany, Sweden and United States. The choice of container material and design is different in the different countries, reflecting differences in the natural conditions, in the fuel and in the size of the operation. These are briefly discussed in the following sub-sections. More details and references are given in the Countries Annex (Annex 1).

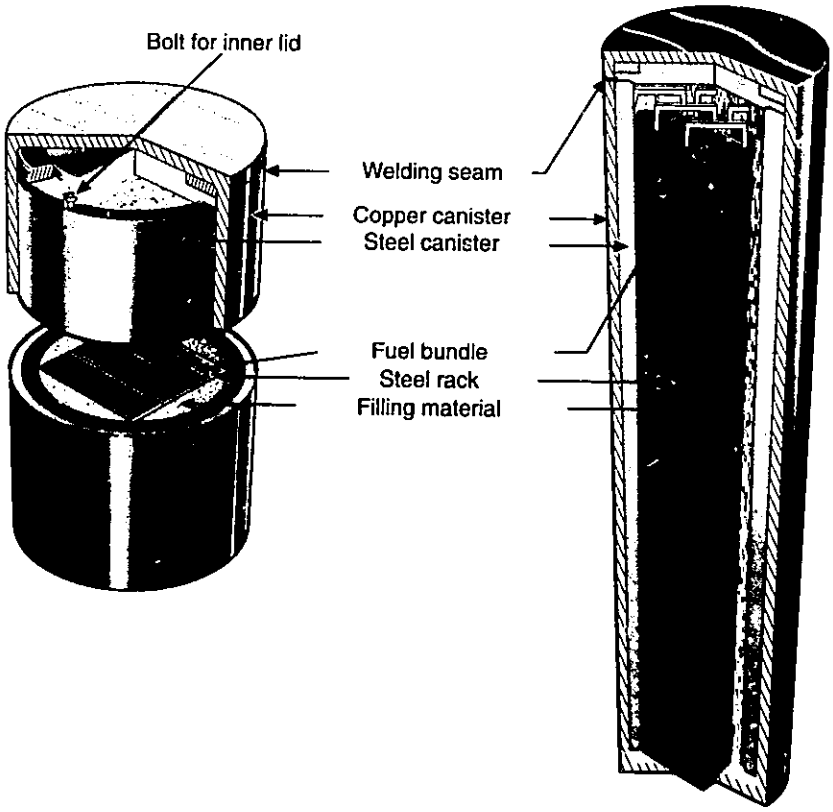
3.2.2.1. Canada

In the Canadian disposal concept, a container fabricated of 6.35-mm-thick grade 2 titanium has been selected as the reference container used in a conceptual engineering study [16]. In the container, 72 irradiated HWR CANDU fuel bundles, corresponding to about 1.4 tU, are packed in a basket consisting of storage pipes. All void space in the container will be filled with a particulate, such as glass beads, that will be vibrationally compacted to a sufficient density to support the container shell against the expected external hydraulic and mechanical loads. The top head will be installed on the container and sealed by diffusion bonding.

The container has been designed to:

- be structurally durable for a period of 500 years after emplacement;
- be amenable to manufacture and inspection, and sparing in its use of non-renewable critical resources required for its manufacture; and
- withstand external pressures of 10 MPa from hydrostatic head and 1 to 3 MPa from the swelling of clay-based sealing materials at a temperature of 100°C.

Figure 3.3 **Canister for spent nuclear fuel (Finnish concept)**



3.2.2.2. Finland

In the Finnish disposal concept, a composite container (Figure 3.3) has been proposed consisting of a 50-mm-thick steel container placed inside a 50-mm copper container [17]. Each container will take 9 BWR fuel assemblies equivalent to 1.6 tU. The void space is planned to be filled with a particulate, *e.g.*, lead shot. The lid of the inner steel container will be bolted, while the lid of the outer copper container will be welded.

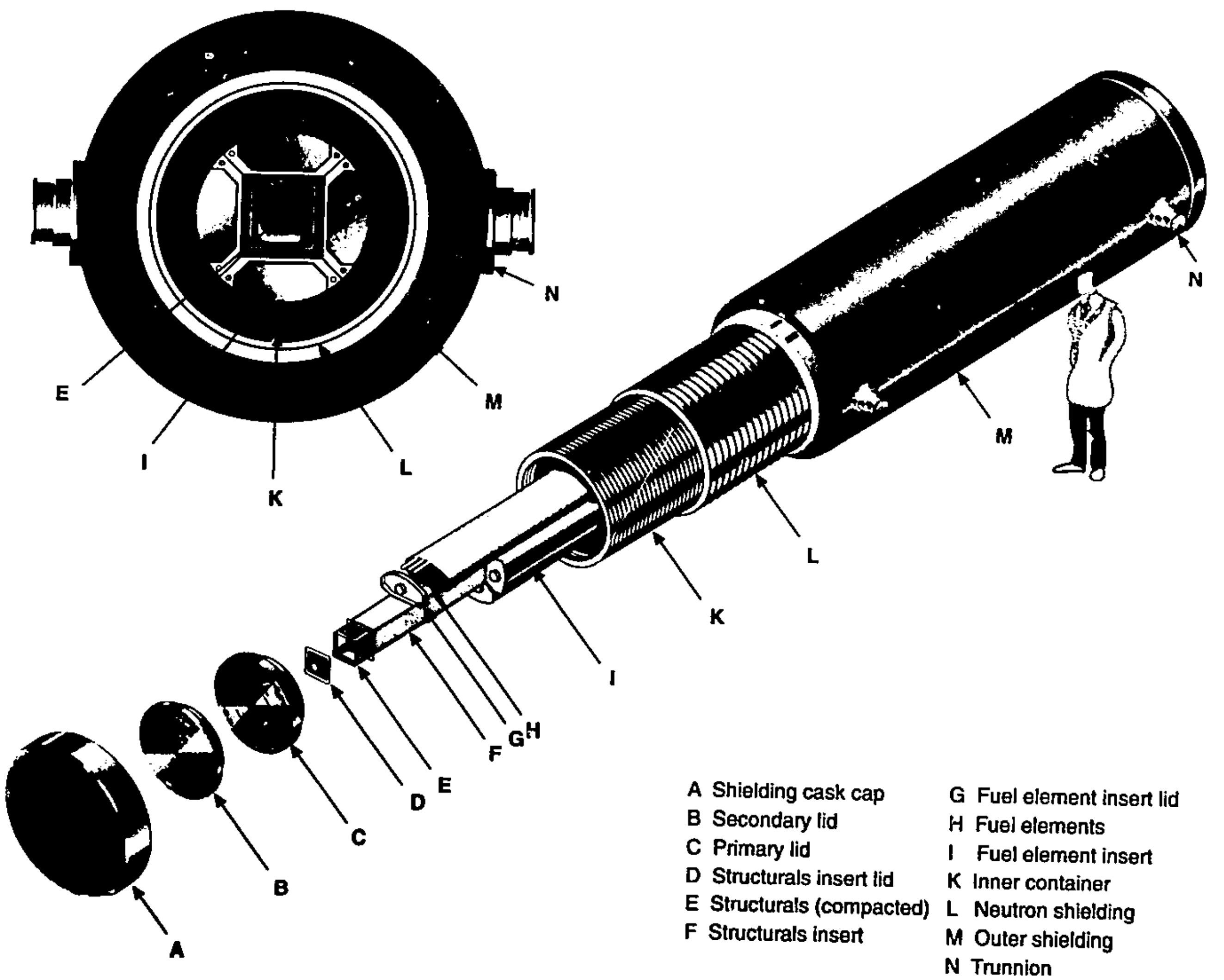
The inner steel container will make the container self-supporting, and the outer copper will give the container a very long service life (corrosion resistance).

3.2.2.3. Germany

In the German disposal concept for spent fuel, the so-called POLLUX cask (Figure 3.4) has been proposed [18]. This consists of a thick-walled, 150-mm, container of reactor steel. Each container can take fuel rods and structural parts from 8 PWR or 24 BWR dismantled fuel assemblies corresponding to 4 tU. The fuel rods are loaded into the inner container and a lid is bolted onto it. Then a second lid is placed on the container and seal-welded. In addition, this container is placed in an outer shielding overpack, thus completing the POLLUX cask.

The container is designed to withstand mechanical stresses due to rock pressure. It also provides adequate shielding during handling and disposal.

Figure 3.4 **Pollux cask**



Credit: Gesellschaft für Nuklear Service (GNS)

3.2.2.4. Spain

The Spanish disposal concepts, developed for two different geological media, granite and salt, consider drift disposal of intact spent fuel packaged in steel containers in a mined repository. The container capacity is 3 PWR or 9 BWR fuel assemblies for the granite option and 4 PWR or 12 BWR fuel assemblies for the salt option.

3.2.2.5. Sweden

In the Swedish disposal concept, a thick-walled, 100-mm copper container has been proposed [19]. The container can accommodate up to nine BWR fuel assemblies or 1.6 tU. After the fuel assemblies are placed in the