Final disposal of high-levels waste and spent nuclear fuel – foreign activities

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SUMMARY

Foreign and international activities on the final disposal of high-level waste and spent nuclear fuel have been reviewed. A considerable research effort is devoted to development of acceptable disposal options. The different technical concepts presently under study are described in the report.

Numerous studies have been made in many countries of the potential risks to future generations from radioactive wastes in underground disposal repositories. In the report the safety assessment studies and existing performance criteria for geological disposal are briefly discussed.

The studies that are being made in Canada, the United States, France and Switzerland are the most interesting for Sweden as these countries also are considering disposal into crystalline rocks.

The overall time-tables in different countries for realisation of the final disposal are rather similar. Normally actual large-scale disposal operations for high-level wastes are not foreseen until after year 2000. In the United States the Congress recently passed the important Nuclear Waste Policy Act. It gives a rather firm timetable for site-selection and construction of nuclear waste disposal facilities. According to this act the first repository for disposal of commercial high-level waste must be in operation not later than January 1998.

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1 INTRODUCTION

In recent years much attention has been focused on final disposal of high-level radioactive wastes, either as spent fuel or as the by-products of nuclear fuel reprocessing. Relatively small volumes of these highly radioactive wastes have been produced so far, but increasing amounts are being generated as the generation of nuclear power continues and grows.

At present there are no operational disposal facilities for these wastes. A considerable research effort is, however, devoted to development of acceptable disposal options.

This report describes national and international research and development programs on disposal of highly radioactive wastes, either as spent nuclear fuel or conditioned high-level waste from reprocessing. There are several reports available describing such programs /1, 2/. The main aim of this report, however, is to put the Swedish R&D-work, the KBS-studies, into a broader international perspective. Therefore, only a brief review of the main national and international programs is given. R&D-strategies, technical concepts and safety assessment including performance criteria will be described in some detail in order to elucidate similarities and differences in concepts and programs.

2 PRESENT SITUATION REGARDING REPROCESSING AND STORAGE OF SPENT FUEL

There are three main options for management of spent fuel:

- reprocessing

- interim storage pending a decision for reprocessing or disposal

- disposal of the fuel as waste in a repository after appropriate conditioning.

The LWR fuel cycle was originally based on the assumption that the spent fuel would be stored at the reactor for one to three years before being reprocessed. Reprocessing plants have been built and operated in some countries. Only in France and the UK, however, such plants are being operated on an industrial scale. Present and planned facilities for reprocessing in some countries is given in Table 1 /3/.

At present, the estimated capacity needed to reprocess the spent fuel generated from existing power reactors should be more than 3 000 tonnes per year. For the year 1990 reprocessing capacities should be about 12 000 tonnes per year. Only a small amount of spent fuel is now being reprocessed. In the coming years there will therefore be a growing accumulation of spent LWR fuel. Forecasts of spent fuel arisings and of storage capacity is given in Figure 1 /3/. They imply that no major problems are foreseen until 1990, whereas special steps have to be taken to solve the problems up to the year 2000. The real situation within an individual country and a particular utility may, however, differ from the regional or global situation.

TABLE I LWR FUEL REPROCESSING FACILITIES IN SOME COUNTRIES

Country	Plant	Owner	Status	Present capacity tHM/a*	Expansion/planned capacity tHM/a* for the period up to 1990
Argentina			Planned		30
Belgium	Dessel-Mol	Government	Reconstruction of former Eurochemic plant		60–300
France	La Hague/UP2	COGEMA	Operational	400	800
	La Hague/UP3	COGEMA	Planned		800
	La Hague/UP3	COGEMA	Planned		800
Germany, Fed. Rep.	Karlsruhe	WAK	Operational	16–35	350
	Commercial plant	DWK	Planned		
India	Tarapur	IAEC	Operational	100	
	Kalpakkam	IAEC	Planned		100
Japan	Tokai	PNC	Operational	210	
	2nd plant	JNFS	Planned		1 200
UK	Windscale	BNFL	Operational	400	
	Windscale (Thorp)	BNFL	Planned		1 200
USA	Barnwell	AGNS	Constructed but not put into operation		

* Tonnes heavy metal per year

Source IAEA Bulletin, Vol.23, No.2 (1981), p.40.

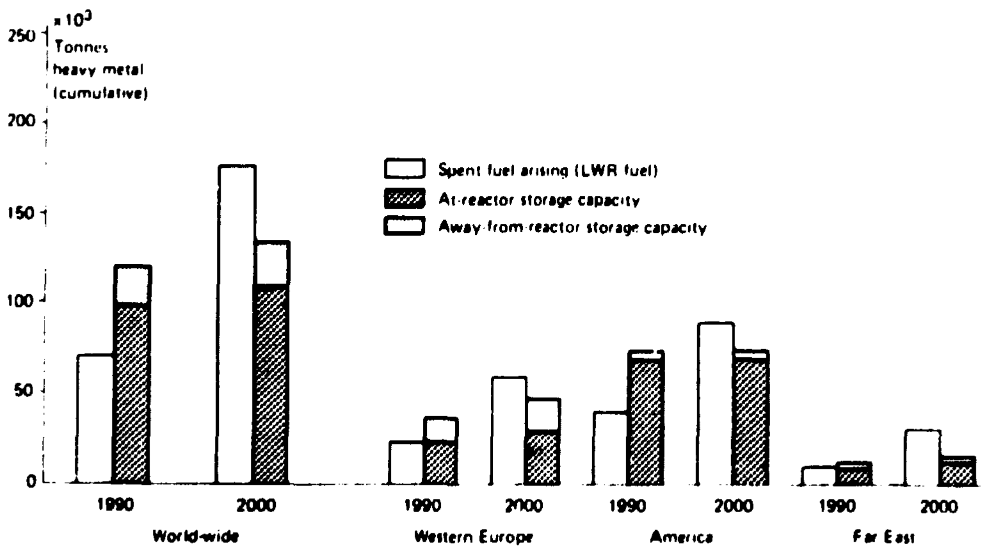

FIG. 1 Forecasts of LWR spent fuel arisings and storage capacity for the year 2000 (figures from CMEA countries not included).

Source: IAEA Bulletin Vol.23, No.2 (1981).

3 ALTERNATIVES FOR THE FINAL DISPOSAL OF HLW AND SPENT NUCLEAR FUEL

3.1 Disposal concepts

The objective of the waste disposal is to prevent the release into the human environment of unacceptable amounts of radionuclides. Different concepts for disposal of high-level waste and spent fuel have been studied or considered /3/:

- seabed disposal
- partitioning and transmutation
- projection into spare
- emplacement in glaciers or ice sheets
- deep geological disposal

3.1.1 Seabed disposal

Large areas of deep ocean sediments are available and most of them are free from seismic and tectonic disruption. A Working Group within the NEA is a forum for coordination and review on the seabed disposal option. Considerable research work is done in the USA on this concept.

The London Convention on Prevention of Marine Pollution by Dumping of Wastes and Other Matter specifically prohibits disposal of high-level waste into the ocean; if seabed disposal is found to be technically feasible and required, changes in international law would be necessary before it could be used.

3.1.2 Partitioning and transmutation

The longlived actinides constitute the main potential radiological health hazard in nuclear wastes after some centuries, because all but a few fission products have by then decayed to insignificant levels. Studies have been carried out in a number of countries on the separation of the actinides from nuclear fuel cycle waste and the subsequent transmutation by nuclear fission in reactors /4, 5, 6/. Some of the conclusions of the various investigations are /7, 8/:

- a large and lengthy development effort would be necessary

- the long-term radiological benefits would be limited and might well be outweighed by the short-term detriment arising from the partitioning-transmutation operations.

3.1.3 Projection into space

It has been proposed to separate and package long-lived waste radionuclides and to place them into solar orbit or to send them into outer space. Accidents during launching are, however, not improbable, at least not in the early stage of the technology.

3.1.4 Emplacement in glaciers or ice sheets

A major uncertainty as regards this option is the long-term global weather pattern affecting the stability of the ice sheets and the sea-level. A great deal of research would be needed to predict future changes.

3.1.5 Deep geological disposal

Emplacement of conditioned waste into deep, stable geological formations is the disposal concept most generally accepted as suitable for high-level waste and spent fuel. The following review is confined to this disposal option.

3.2 Alternatives for deep geological disposal

The following geological formations are currently studied for disposal of high-level waste:

- crystalline rocks (e.g. granite and gneiss)
- rock salt (bedded formations and salt domes)
- clays
- basalt
- volcanic tuff.

3.2.1 Crystalline rocks

Crystalline rocks are generally very impermeable to water. The only significant pathway by which nuclides could return to the biosphere from a repository is by dissolution in ground water and migration through fracture networks. The crystalline formations have great structural strength and resistance to erosion and other disruptive events.

Decay heat from the radioactive waste raises the temperature in the surrounding rock. The temperature distribution can create stresses in the rock with possible consequences of slow crack growth and creep near the repository. The heat may also influence local ground water flow.

Disposal in crystalline rocks is considered in many countries: Canada, Finland, Japan, Spain, Sweden, Switzerland, United Kingdom, USA.

3.2.2 Clay formations

The clay minerals are composed of hydrous aluminium silicates and the particle size is often less than 0.1 mm. The clay formations are very impermeable and act as an effective barrier to ground water movement. They sorb most radionuclides very effectively. Clays are self-healing for cracks and fractures owing to their plasticity. They are, however, sensitive to heat and gamma-radiation owing to the presence of interstitial water. Clay is easy to work, but requires structural support of tunnels and openings.

The clay-formation option is studied mainly in Belgium and Italy.

3.2.3 Salt formations

Salt deposits exist either as bedded salt formations or as salt domes. The latter are extrusions from deeper bedded layers which have risen to near the surface under the influence of lithostatic pressure and density differences. Undisturbed formations are essentially impermeable and have a low water content. The high thermal conductivity gives good heat dissipation. Its plasticity, which increases with temperature, heals cracks and will seal in the waste as the salt backfill consolidates. Radionuclide migration in ground water can only be envisaged if the formation is disrupted or subjected to dissolution in abundant fresh water. Although there is no ground water flow in salt in an undisturbed situation, the waste may come into contact with a limited amount of brine from its migration up the temperature gradient around the high-level waste. Disposal in salt formations is studied in many countries: USA, Germany, Netherlands, Denmark, Spain.

3.2.4 Basalt

Basalt is a fine-grained, dark, volcanic rock formed from lava flows. Formations of basalt are fractured in a regular joint pattern through which ground water moves. The basalt alternative is extensively studied in the USA (the Hanford area).

3.2.5 Tuff

Tuff is volcanic ash which has been compressed under its own weight and sometimes welded owing to its high temperature on deposition. The permeability of the more consolidated tuffs is low and their sorptive capacities high. Tuff formations in Nevada, USA, are investigated as an alternative for siting of a repository for high-level waste. Significant tuff deposits also exist in Italy.

4. GENERAL OVERVIEW

Brief program overviews of fuel cycle, spent fuel and waste management activities were presented by staff members of Pacific Northwest Laborabory in 1982 for countries and international organizations with substantial programs in these areas /2/. Table 2 summarizes the fuel cycle strategy of some of the selected countries.

TABLE 2 Fuel Cycle Strategy

Country	Fuel Cycle	Spent Fuel Disposition	HLW Management
Belgium	U-Pu (LWR)	Reprocess and recycle Pu to LWR or FBR. Interim--foreign reprocessing; long term--domestic.	Vitrify HLW (French process) and isolate in clay repository.
Canada	U-Pu (CANDU PHWR); conversion to ^{233}U-Th PHWR is being evaluated.	Current--long term retrievable storage; long term--geologic disposal or domestic reprocessing.	Isolate spent fuel or HLW in granite repository.
Finland	U-Pu (LWR)	Russian fuels--return to Russia. Swedish fuels--geologic disposal or foreign reprocessing.	Isolate non-Russian spent fuel or HLW in granite repository.
France	U-Pu (GCR, LWR, FBR)	Reprocess (domestic) and recycle Pu to FBRs.	Vitrify HLW; provide long-term interim storage; isolate in granite repository.
Germany (FRG)	U-Pu (LWR, FBR, HTGR)	AFR storage (dry); reprocess (interim--foreign; long-term--domestic) and recycle Pu to LWR or FBR.	Vitrify HLW (French process or German ceramic melter technology) and isolate in salt dome repository.
Japan	U-Pu (LWR, HWR, FBR)	Reprocess (foreign and domestic) and recycle Pu to HWR and FBR.	Vitrify (ceramic melter process); long-term storage geologic disposal.
Netherlands	U-Pu (LWR)	Foreign reprocessing.	Isolate HLW glass in salt dome repository.

TABLE 2 (contd)

Country	Fuel Cycle	Spent Fuel Disposition	HLW Management
Sweden	U-Pu (LWR)	AFR storage; spent fuel disposal or foreign reprocessing.	Isolate spent fuel or HLW glass in granite repository.
Switzerland	U-Pu (LWR)	Foreign reprocessing.	Isolate HLW glass in granite formation.
USSR	U-Pu (LWR, FBR, LGR)	Interim; pool storage; long-term: Domestic reprocessing.	Vitrify HLW and isolate in geologic repository.
United Kingdom	U-Pu (GCR, FBR)	Domestic reprocessing; recycle Pu to FBR.	Vitrify HLW (French process); provide long-term storage; delay decision on terminal disposal.
USA	U-Pu (LWR, FBR, HTGR)	Domestic reprocessing.	Vitrify HLW (ceramic melter); isolate in geologic repository (salt, basalt, tuff).

5 BRIEF REVIEWS OF SOME NATIONAL AND INTERNATIONAL PROGRAMMES

5.1 The United States

At the end of 1982 the installed nuclear capacity in the United States was about 64 GWe and 74 GWe was under construction /9/. Most of the reactors are light water moderated.

To date most of the spent fuel elements have been stored in pools near the reactors and the low-level waste has been disposed of in shallow trenches. The military programme has generated large volumes of reprocessing waste such as liquid and solid high-level waste and low-level transuranic waste. Most of this waste is waiting for final conditioning and disposal, and much of the research work in this field is aimed at the solution of these problems.

According to the original plans, the spent fuel was to be reprocessed for plutonium recovery, but in 1977 this intention was indefinitely deferred by the Carter administration for fear of the proliferation of nuclear weapons.

The Nuclear Fuel Services Plant in New York State operated 1966-1972 but is now shut down. Most high-activity wastes from this plant were made alkaline and are stored in mild-steel tanks but small quantities of special wastes are stored in a stainless-steel tank. The future of the Barnwell Nuclear Services Plant in South Carolina is under discussion.

The overall strategy for disposal of high-level radioactive waste is to develop mined repositories in stable geological formations, while continuing to examine the feasibility of subseabed disposal as a complement and longer-ranged option /10, 11/.

Alternative disposal methods, like e.g. emplacement in very deep drillholes, ejection into space, transmutation, or emplacement in a mined cavity that leads to rock melting, have been seriously considered but discarded.

The management of civil nuclear waste is governed by federal laws. The Atomic Energy Act of 1954 was followed by the Energy Reorganization Act of 1974, which replaced the Atomic Energy Commission, AEC, with two new organizations. These were the Energy Research and Development Administration, ERDA, and the Nuclear Regulatory Commission, NRC. The Environmental Protection Agency, EPA, was also given responsibility for these matters. The U S Department of Energy Organization Act of 1977 rendered ERDA superfluous and this work is now directed by the Department of Energy, DOE.

The three organizations, DOE, NRC and EPA, now share the major federal responsibility for developing an adequate solution to the disposal of high-level waste. DOE has management responsibility for defence nuclear waste, commercial spent fuel assigned to DOE, and commercial high-level nuclear waste in federal custody. DOE also has responsibility for ensuring that transportation systems exist to support federal programmes.

The repositories will be federally licensed and owned.

NRC is responsible for regulation of the DOE's waste disposal activities with primary emphasis on safety and environmental acceptability. This regulatory power is exercised through licensing of the facilities for civil nuclear waste.

EPA is responsible for the determination of criteria and standards for the disposal of radioactive waste, which are to be implemented by NRC.

In March 1978, President Carter established the Interagency Review Group on Radioactive Waste Management, IRG. Based on their report, President Carter, in February 1980, called for a National Plan for Radioactive Waste Management, NPRWM, which would include all the elements involved in radioactive waste management (i.e. high-level waste, transuranic waste, low-level waste, mill tailings, decomissioning, transportation, public participation, regulation, etc). This programme put the emphasis on mined geological repositories for the disposal of high-level and transuranic waste. The local states will participate in the site selection process in accordance with the principle of consultation and concurrence. A State Planning Council, SPC, including state representatives and heads of federal agencies has been created to facilitate co-operation in these and similar matters.

The development programme for the long-term management of high-level waste was organized by DOE in the National Waste Terminal Storage Program, NWTS /12, 13, 14/.

Solidified wastes from reprocessed fuel will be used as the reference form for disposal, however, disposal of spent fuel will be kept as an option, should the utilities choose not to reprocess it.

Three principal geological formations are currently under investigation: basalt (on the Hanford Site); tuff (on the Nevada Test Site); bedded salt (the Paradox Basin in Utah) and salt domes (in Texas, Louisiana and Mississippi).

Regional surveys in crystalline rock areas were started in 1981. Such rocks occur in southern, north-central and north-eastern areas of the US. A site for more detailed investigations will be identified during 1980´s. The Climax Mine in a granite formation on the Nevada Test Site is used for rock mechanics and tracer migration studies in fractured granite. A few canisters of commercial light water reactor spent fuel and additional heater simulators are placed in boreholes beneath tunnels to study temperature distributions and stress in granite, and to investigate any effects associated with the intense radiation from spent fuel.

In December 1982 Congress passed the Nuclear Waste Policy Act /15/. It was signed into law by President Reagan January 7, 1983. According to this law DOE must recommend three out of five sites to the President for consideration by Jan. 1, 1985. The President has 60 days to approve the recommendations for site characterization, which primarily involves sinking a shaft on the site. DOE must submit a plan to the state involved and hold public hearings before actually sinking the shaft and again before recommending a single site to the President. The President must recommend a single site for the first repository to Congress by March 31, 1987, a date which can be extended by up to 12 months.

The time schedule is similar for the second repository site, with the first recommendation to the President of three sites by July 1, 1989, and the President's final recommendation to Congress by March 31, 1990. DOE must make its choice of three sites from five candidates, at least three of which were not included in the first round. If a site is not chosen initially for characterization, it must be excluded from consideration.

Once the site is approved, DOE within 90 days must go to NRC for a construction permit, and NRC has by Jan. 1, 1989 or within three years of submission to approve the application for the first site, a date which can be extended by one year. The equivalent date for the second repository site is Jan. 1, 1992, or within three years of submission, and with a one-year extension possible.

The first repository must be in operation beginning not later than January 31, 1998.

The Nuclear Waste Policy Act of 1982 also calls for a recommendation by DOE on the first site for a monitored retrievable storage (MRS) within two-and-a-half years. Congress cannot site an MRS facility in a state which is a candidate for or already has a permanent repository, but Congress can transform an MRS site into a permanent repository. A monitored retrievable storage could fullfill three missions:

- interim storage of spent fuel before reprocessing

- storage of spent fuel or reprocessed HLW before disposal

- indefinite storage (disposal) of spent fuel or HLW /16/.

5.2 Canada

Canada has developed a nuclear power capacity based on the domestic CANDU pressurized heavy water reactor system, with one exception (Gentilly, heavy water boiling reactor). All planned reactors are of the CANDU type. At the end of 1982 14 units (total 7 700 MWe) were in operation and 12 reactors (total 8 800 MWe) under construction /9/.

Spent fuel is stored in pools at the utility sites and a centralized interim storage facility at a federal Fuel Cycle Center is contemplated. This might be owned and operated either by the utility or by the federal government.

Atomic Energy of Canada Limited, AECL, is a federal agency, responsible for the nuclear energy programme. In 1978, AECL was reorganized into a holding company with four subsidiaries. One of these is AECL Research Company, which is reponsible for the Chalk River and Whiteshell research laboratories. Research and development in the field of radioactive waste management is done at Whiteshell except as regards reactor waste, which has been assigned to Chalk River.

A Technical Advisory Committee, TAC, was set up in mid-1979 to advise AECL on the extent and quality of the technical programme on nuclear fuel waste management, acting as an independent review committee.

DFE, Department for Environment, has general responsibility for environmental protection, but

DFE has also been contracted by AECL for hydrogeological and hydrogeochemical work in connection with the search for a geological waste repository. The geological investigations in this field are done for AECL by the Geologic Survey of Canada, which is a branch of Department of Energy, Mines and Resources, EMR. The Mining Research Laboratories of Ottawa are also a part of EMR being involved in waste deposition research.

AECL has initiated an elaborate study under the name Nuclear Fuel Waste Management Program. No choice has been made on whether irradiated fuel or conditioned high-level waste will be eventually disposed of in geological formations. Research takes both possibilities into consideration.

Research activities have focused on deep underground disposal in large igneous intrusions of crystalline rock, known as plutons /17, 13/. Other media including salt, limestone, and shale are considered possible alternatives, but work on those is very limited.

An underground research laboratory in a granite pluton will be constructed beginning in 1983, in an area near the Whiteshell Nuclear Research Establishment in Manitoba /19/. The facility, URL, will be used for basic research, and no nuclear fuel waste will be placed at this site. The generic research phase of the program will continue until 1991. Operation of commercial disposal vaults is neither expected, nor required, until well after the year 2000 /1/.

5.3 The United Kingdom

Nuclear power reactors with a capacity of 9.7 GWe were in operation at the end of 1982 and about 3.7 GWe under construction. The first reactors in the United Kingdom were gas-cooled graphite moderated reactors (Magnox). These have been followed by advanced gas-cooled reactors (AGR). Light water reactors are considered for the future.

Magnox fuel and oxide fuel from AGR´s were treated at the Windscale (Sellafield) reprocessing plant during the period 1964-73, and a new reprocessing plant (THORP 1200 tonnes/year) for oxide fuel will be constructed at Sellafield.

Liquid wastes are stored as acidic solutions at Sellafield and Dounreay (fast reactor fuel) reprocessing plants in stainless steel tanks /1/. It is planned from the late 1980´s to vitrify the waste using the French AVM system. The vitrified blocks will be placed in a specially designed store, cooled by air or water, on or near the surface for at least 50 years /20/.

The production of nuclear fuel and reprocessing is handled by British Nuclear Fuel Limited, BNFL, which is also active in the field of research and development for waste management. The energy producing facilities are usually owned by the Central Electricity Generating Board, CEGB, but reactors situated in Scotland by the South of Scotland Electricity Board, SSEB.

Research and development on waste conditioning and disposal is to a large extent covered by The United Kingdom Atomic Energy Authority, UKAEA, e.g. vitrification of high-level reprocessing waste.

The National Radiological Protection Board, NPRB, was formed on the basis of the Radiological Protection Act, 1970, to investigate the radiation hazards and to advise authorities concering radiation protections in England. NRPB is playing an active part in the safety analyses for radioactive waste disposal.

The possibilities for disposal being considered are placing the vitrified high-level waste on or under the bed of the ocean or in deep geological formations on land. The full operation of a disposal facility for high-level waste is not judged to be necessary before 2040 /1/.

Exploratory drilling into granite has been performed at one site in northern Scotland and a programme of field studies of heat transfer, fracture hydrology, and radionuclide transfer is in progress (at the AERE test site on Cornish granite).

Research will continue in order to help determine eventually which is the best option in terms of safety, environmental acceptability, flexibility and costs, plus the requirements placed on future generations. This will not, for the time being, require further fieldwork in the UK according to a paper, presented to Parliament by the government in 1982 /21, 22/.

The government will, however, follow closely work in other countries and, where appropriate, will join in international collaborative projects on land disposal as well as ocean disposal.

5.4 France

In France most of the installed nuclear power capacity (25 GWe at the end of 1982) is pressurized water reactors. Several PWRs and a large liquid metal fast breeder are under construction (total capacity 32 GWe at the end of 1982) /9/. The nuclear power plants are, with a few exceptions, operated by the government-owned company Electricité de France, EDF.

Commissariat à l´Energie Atomique, CEA, is the French Atomic Energy Commission under the direct authority of the Minstry of Industry and Research. CEA is responsible for controlling all nuclear activities including the military part. All activities in the field of nuclear energy are effectively directed by the government.

Companie Générale des Matières Nucléaires, COGEMA, which is owned by the government, was created in 1976 to take care of the commercial aspects of the nuclear fuel cycle such as uranium mining, exploration, enrichment, reprocessing (La Hague and Marcoule), fuel fabrication and transportation.

Société Générale pour les Techniques Nouvelles, SGN, earlier Saint Gobain Techniques Nouvelles, acts as a subsidiary to COGEMA and is responsible for an extensive programme of construction and development. Important areas are plants for fuel fabrication, reprocessing, waste conditioning and pool storage. Laboratory equipment for radioactive work is also being manufactured.

In order to treat the problem of radioactive waste disposal on a more industrial basis, a new organization, l´Agence National pour le Gestion des Déchets Radioactifs, ANDRA, has been created within CEA.

ANDRA has been given the following responsibilities:

- to arrange for the long-term storage of waste,
- to issue conditioning and storage regulations in co-operation with nuclear energy producers,
- to establish new centres for long-term storage,
- to contribute to research and development concerning long-term storage.

Institut pour la Protection et la Sureté Nucléaire, IPSN, is the special branch of CEA for all technical safety problems concerning nuclear facilities (reactor safety, nuclear waste handling and disposal, decommissioning, etc).

The official policy for safety in the nuclear industry is established by an interministerial committee, Conseil Supérieur de Sureté Nucléaire. A working group, attached to the Council and chaired by Raimond Castaing, issued in December 1982 a report on reprocessing and waste management /23/. The group recommends, among other things, that:

- study should be made of the technique of advanced reprocessing with the separation of minor actinides, with a determination to do everything possible to ensure that it is effectively applied before the end of the centrury.

This technique, developed by CEA on laboratory scale, is designed to strip from the wastes the very long half-life alpha emitters and other elements capable of producing them as daughter products,

- investigations should be initiated immediately to acquire industrial know-how on options other than immediate reprocessing, including the definitive storage of spent fuels.

Liquid wastes are now stored as acidic solutions at the Marcoule and La Hague reprocessing plants in stainless steel tanks. The AVM plant (Marcoule) commenced to solidify wastes into borosilicate glass in 1978. Solidified waste will be stored in air-cooled vaults. A similar vitrifications plant will be installed at La Hague, scheduled for operation by 1987 /24, 25/.

Several types of geological formations are considered for disposal of high-level wastes: granite, salt, clay /26/. The investigation program has started with granite formations in the Massif Central. The objective is to choose a site for construction of a deep experimental cavity, in which the actual conditions of storage, geological characteristics, hydrogeology and thermomechanical properties can be studied. ANDRA is planning to have the demonstration disposal center for vitrified wastes operational around 1993 /25/.

5.5 The Federal Republic of Germany

Nuclear power production is based on light-water reactors with a current capacity of 10 GWe

(18 GWe under construction). Germany has a reprocessing contract with COGEMA in France for about 2 300 tonnes of uranium but this is not sufficient for the whole production of spent fuel. Therefore, plans have been made to develop domestic reprocessing capacity and also to provide for interim storage capacity.

The Federal Republic of Germany has a unique concept for the interim storage of spent fuel. Fuel elements will be stored in special, steel transport containers (the STEAG Castor container), cooled by natural convection /27/. Two central storage facilities for spent fuel using these containers are planned in Gorleben and Aahaus. Each storage facility will have a capacity of 1 500 tU.

These storage facilities are considered to be more flexible and convenient for German conditions than conventional wet storage.

The original plans for a large centre for reprocessing (1 400 tonnes/year) and MOX-fuel fabrication in Gorleben in Lower Saxony have been delayed for political reasons. However, the investigation of the salt dome in Gorleben to check its suitability as a waste repository is being continued as is the licensing procedure for an interim storage of spent fuel at the site.

Deutsche Gesellschaft für Wiederaufarbeitung von Kernbrennstoffen mbH, DWK, is an utility-owned company, which was created in connection with the Gorleben project to build and operate the reprocessing plants in Germany. DWK acquired Gesellschaft für Wiederaufarbeitung von Kernbrennstoffen mbH, GWK, which is running a demonstration reprocessing plant in Karlsruhe (40 tonnes/year), and Kernbrennstoff-Wiederaufarbeitungsgesellschaft mbH, KEWA, which is an equal sharehold together with BNFL (United Kingdom) and COGEMA (France) in United Reprocessors GmbH. This was done to concentrate efforts in the field of reprocessing.

In Germany, three different processes for the solidification of high-level waste solutions from reprocessing have been developed: VERA, at Karlsruhe, FIPS at Jülich and PAMELA, developed by the Gelsenberg Company and Eurochemic. Since 1977, the work has been concentrated to the PAMELA process and this will be used for high-level waste solution from the reprocessing of oxide fuel, being stored at the Eurochemic plant in Belgium in accordance with an agreement between Eurochemic and DWK. It is planned that this plant will come into operation between 1985 and 1988.

The DWK company in Germany has chosen the French AVM process as a reference design and an AVM plant, known as HOVA, will be built at the Karlsruhe reprocessing plant, WAK, and come into operation in 1986.

The main objective of the German waste management programme is to reprocess spent fuel and isolate the waste in mined salt dome repositories after conditioning of the waste, but alternative spent fuel management and disposal techniques are studied in a special programme /28/. A variety of concepts for containers, conditioning procedures, and disposal options are investigated. An assessment study, in which the waste management including reprocessing is compared with direct disposal of spent fuel, is scheduled for the end of 1984.

5.6 Switzerland

The four light water reactors in Switzerland have a joint capacity of nearly 2 GWe and a fifth reactor is under construction. Spent fuel will be reprocessed in La Hague in accordance with a contract with COGEMA, which covers spent fuel through 1989. The vitrified high-level waste will be sent back for final disposal in Switzerland.

In a referendum held in May 1979, an amendment to the Atomic Act (Atomgesetz) from 1959 was passed. It is now necessary for a producer of radioative waste to take care of the safe and final disposal of radioactive waste. Moreover, in order to receive permission to operate existing nuclear power plants, the utilities must present an acceptable disposal plan before December 31, 1985, showing how and where the waste can be disposed of in a safe way.

The producers of radioactive waste, that is the nuclear power utilities and the Swiss Government, have jointly founded the company Nationale Genossenschaft für die Lagerung radioaktiver Abfälle, NAGRA.

NAGRA was founded as early as 1972 and its task was then to find a subsurface repository for low- and medium-active waste. The objective is now to meet the conditions of the amended Atomic Act.

Switzerland favours the option of geological disposal in hard rock. High-level waste repositories will probably be developed in deep granite formations overlain by thick layers of sedimentary rocks, and an ambitious test drilling programme has been started by NAGRA. An in situ test station will be constructed (Felslabor Grimsel) /29/ in the near future.

5.7 Japan

The great dependence on other nations for its energy supplies has led Japan to adopt a very ambitious nuclear power programme. The present nuclear capacity is about 17 GWe and 18 GWe is under construction.

After a reorganization in January 1979, safety regulation of reactors from construction permission to decommissioning is to be implemented by the Ministry of International Trade and Industry, MITI, the Ministry of Transport, MoT, and the Science and Technology Agency, STA. The safety regulations for fuel facilitites such as reprocessing plants, fuel fabrication plants and other facilities involved in the treatment and production of nuclear fuel, are to be implemented by STA.

Research and development in the field of radioactive waste management is, to a large part, conducted by STA-controlled organizations such as JAERI (Japan Atomic Energy Research Institute), PNC (Power Reactor and Nuclear Fuel Development Corportion) and different national research institutes.

A reprocessing plant at Tokai Mura commenced operation in 1977 and liquid wastes are stored in stainless steel tanks /30/. Solidification processes are being developed and a pilot plant will be constructed.

The capacity of the Tokai Reprocessing Plant (0.7 t/d) is not enough for processing all spent fuels discharged from Japanese power stations. While waiting for the operation of the second commercial plant of 5 ton/day capacity in 1990, 3 200 ton of spent fuel are being sent to the plants of British Nuclear Fuels Limited and of French COGEMA for reprocessing /30/.

The geological disposal program is planned to be conducted in stages as indicated in Figure 2 /31, 32/. Survey of potential geological formations by 1984; research on candidate geological formations by 1995; in-situ test with simulated waste; in-situ test with actual waste and trial disposal by 2015.

Geological formations under consideration are granite, diabase, shale, zeolitic tuff, limestone and schist.

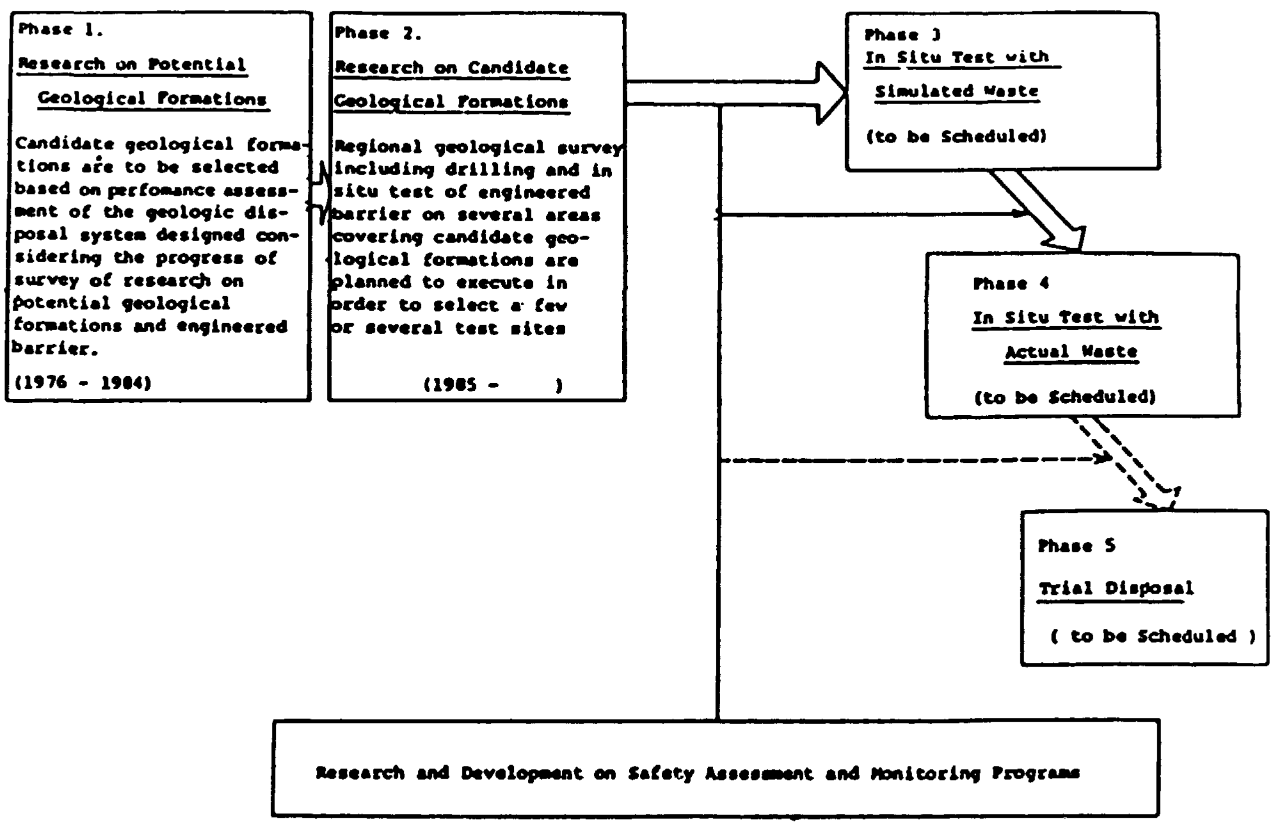

FIGURE 2 General Scheme for the Geological Disposal Program in Japan.

5.8 Belgium

Six power reactors with a capacity of 3.6 GWe are in operation in Belgium and two (2.0 GWe) are under construction.

Belgian utilities have reprocessing contracts with COGEMA totaling 540 MTU /33/.

Liquid wastes from the Eurochemic reprocessing plant at Mol are stored in stainless steel tanks. Vitrifications processes are being considered for waste solidification, including the incorporation of a granular product into a metallic matrix. Solidified waste will be placed in engineered surface storage /1/.

Plastic clay has been selected as first choice formation for the research and development program on the disposal of conditioned high level and alpha-bearing wastes. The investigations have concentrated on a clay formation underlying the area around the nuclear energy research centre at Mol (the Boom clay formation). An access shaft has been drilled into the clay to a depth of - 227 m and at its lower end a horizontal gallery is being constructed. It will essentially serve for geotechnical studies in-situ, experiments on heat dissipation and corrosion, tests on backfilling materials and methods, and finally, test on the behaviour of waste materials. It is expected that data required for the licensing procedure for the construction of an operational repository will be available around 1987-88 /34/.

5.9 The Soviet Union

The Soviet Union has an installed nuclear power capacity of more than 17 GWe and 24 GWe under construction. Most of the present units are either light-water cooled graphite-moderated reactors or pressurized water reactors. All aspects of the fuel cycle are being considered and developed within the country including breeders, reprocessing and enrichment plants. There are also advanced plans for a nuclear district heating system.

Spent fuel is supposed to be reprocessed /35/. Liquid wastes are stored in stainless-steel tanks. Solidification processes to produce phosphate and borosilicate glasses have been investigated on a laboratory scale with radioactive wastes and on a pilot plant scale with inactive simulated wastes.

Industrial-scale plant to vitrify wastes is expected to begin operation in the 1980´s. Storage of solidified wastes in near-surface facilities and deep geological disposal concepts are being studied /1/. No details, however, are available.

5.10 Spain

About 2 GWe nuclear power is in operation and 9 GWe under construction. JEN (Junta de Energia Nuclear) is charged with research and with storage of radioactive waste. For terminal storage of high-level wastes salt formations have been considered, but granite formations are also an option. A facility for terminal storage of high-level waste will not be necessary before the year 2000.

5.11 Finland

Four light-water moderated reactors with a capacity of 2.2 GWe are in operation in Finland. Spent fuel from the two Soviet-build Loviisa reactors are sent to the USSR. For the other two reactors, build by the Swedish ASEA-ATOM, a disposal option is needed. Crystalline rock formations are considered for the disposal of spent fuel from these reactors. Studies are in progress on site selection and on canister corrosion, leaching properties of high-level waste and radionuclide migration. A safety analysis of disposal of spent fuel was published in 1982 /36/.

5.12 Some international activities

5.12.1 The Stripa Project

The International Stripa Project is supported by several countries of the OECD Nuclear Energy Agency (NEA): Canada, Finland, France, Japan, Sweden, Switzerland, UK and USA.

The Stripa mine is an abandoned iron ore mine in Sweden.A granite formation is adjacent to the ore excavations and is accessible at a depth of 350 meters.

Research concerns geochemical and hydrological studies, engineered barriers and sealing of boreholes, tunnels and shafts, geophysical methods and migration experiments.

5.12.2 Seabed disposal

Seabed disposal of high-level waste would involve the placement of conditioned and packaged waste into geologically stable clay sediments. A seabed Working Group within the NEA is a forum for coordination and review of research work on the seabed disposal option /1/. Belgium, Canada, the Federal Republic of Germany, France, Japan, the Netherlands, Switzerland, the United Kingdom and the USA are represented in the group.

5.12.3 ISIRS (International Sorption Information Retrieval System)

The International Sorption Information Retrieval System has been set up within NEA to develop a computer-based data storage system for the results of radionuclide sorption experiments.

Eleven NEA countries support this project: Canada, Finland, France, the Federal Republic of Germany, Italy, Japan, the Netherlands, Sweden, Switzerland, the United Kingdom and the United States. The project is directed by a technical committee of representatives from participating countries. For the initial two-year period, the project is based at the Battelle Pacific Northwest Laboratory in Richland, USA. During 1981, computer software systems have been adapted to the needs of the international community, and specifications for a common reporting format have been elaborated.

During 1982 the system will become available for selective retrieval of data from laboratories throughout the participating countries.

5.12.4 The radioactive waste management program of the Commission of the European Communities

For several years, the Commission of the European Communities has been supporting various research activities conducted in the member countries as well as by its Joint Research Centre in the field of radioactive waste management and disposal. The present multi-annual R & D programme is running over the period 1980-1983 and in part trough 1984.

The Performance Assessment of Geological Isolation Systems (PAGIS) project is an evaluation of the confinement performances and a risk analysis of various disposal systems carried out by the Commission of European Communities (CEC) Joint Research Centre and by research bodies within the European community under CEC coordination and partial financing /37/. The study will develop in three phases: collection of data and definitions of the study, implementation, and evaluation and conclusion. Preliminary results are expected in 1984-85.

The interaction of radionuclides with the environment is studied in a coordinated activity involving several national laboratories (The MIRAGE project). The behaviour of container materials (Ti-alloys, Hastelloy, mild steel) is also investigated by several laboratories.

5.12.5 The IAEA-activities in waste disposal

The IAEA has been concerned with radioactive waste management since its inception. The objectives of the Agency's wastes management programme are to assist its Member States in the safe and effective management of wastes by organizing the exchange and dissemination of information, providing guidance and technical assistance and supporting research.

As waste disposal is the current issue of highest interest, an Agency programme was set up in 1977 to develop a set of guidelines on the safe underground disposal of low-, intermediate- and high-level wastes in shallow ground, rock cavities or deep geological repositories. This programme will continue until 1990. Eleven Safety Series and Technical documents and reports have been published under this programme so far, which also addresses safety and other criteria for waste disposal.

6 TECHNICAL CONCEPTS FOR FINAL DISPOSAL

6.1 Waste forms

6.1.1 High-level waste from reprocessing

In the US many candidate waste forms have been investigated as potential media for the immobilization and geologic disposal of HLW resulting from chemical processing of nuclear reactor fuels: high-silica glass, borosilicate glass, SYNROC, tailored ceramic, concrete, coated sol-gel particles, glass marbles in lead-matrix. Two of these HLW forms have been selected for intense development /38/. Borosilicate glass is the reference form. A crystalline ceramic waste form, SYNROC, has been chosen for further product formulation and process development as the alternative to borosilicate glass.

In France high-level liquid wastes from reprocessing is solidified into borosilicate glass at the AVM plant (Marcoule). The French AVM system is planned to be used in UK and in the Federal Republic of Germany. Borosilicate glass will also be the product of the Japanese pilot plant.

The PAMELA process, which may produce glass beads in a lead matrix, will be used for high-level waste solutions stored at the Eurochemic plant at Mol in Belgium.

6.1.2 Spent fuel

Spent fuel can be packaged for disposal as unmodified assemblies, but several alternatives to this reference case have been studied in the US /39/:

Alternative 1 - end-fitting removal
2 - fission-gas venting and resealing
3 - fuel disassembly and close-packing of fuel pins
4 - fuel shearing and immobilization in a solid matrix

An assessment of the in-repository performance of the alternative waste forms has been conducted. It was concluded that, although there is insufficient information to rule out any of the waste forms, the most unfavorable alternative is the sheared/immobilized waste form.

Similar studies are in progress in the Federal Republic of Germany /28, 40, 41/. The following cases are being investigated:

- the unmodified fuel element
- fuel elements without head and foot pieces
- dissassembled spent fuel element (rods)
- degassed fuel rods
- shortened fuel rods (i.e. folded rods, rods cut to 1 m, rods cut to 0.05 m pieces)
- voloxidation and vitrification of fuel material
- dissolution and vitrification of fuel material.

Of these alternatives the following have been selected for more intense development /28/:

- the unmodified fuel element
- dissassembled spent fuel (rods)

- shortened fuel rods.

The product will be a gas-tight container that holds the spent fuel and that is suitable for longer-term storage in a draft air-cooled vault storage system. The filling of the void volume of the dry storage container with a suitable matrix is also considered. The final conditioning is done by emplacing the intermediate product into the properly constructed final disposal cask and sealing the cask. One is a multilayer cask, where the inner layer from a cheap metallic construction material gives the necessary container stability, whereas the outer layers, constructed from more expensive high-quality materials, provide corrosion resistance against salt brine at elevated temperatures /41/.

The German program on spent fuel will be completed in 1984 with an evaluation of the different alternatives.

Also in Canada spent fuel is one alternative waste form for disposal in geological formations.

6.2 Container materials

Container concepts under investigation include metals, graphite and ceramic materials.

In the US initial screening studies of corrosion resistent structural components in both brines and basaltic ground water have given as result that titanium and the alloy Ti Code-12 have been selected as primary candidate metallic materials for a thin overpack. Backup materials include the nickel-based alloys. For the self-shielded concept, using a single thick structural barrier, studies will be carried out on the cast irons and cast steel (42, 43). Limited studies have also been performed on graphite and ceramic materials.

In Canada potential container-shell materials have been evaluated from the viewpoints of corrosion resistance, ease of fabrication and mechanical properties. The following main reference materials have been recommended: Type 316L stainless steel, high nickelbase alloys, dilute titanium-base alloys and copper (44, 45).

In the UK the current research for disposal container material is concentrating on metallic materials (46). Two concepts are being pursued involving the use of metals possessing high corrosion resistance, and also metals of comparatively low resistance, which can be used economically at considerable thickness to make allowance for corrosion losses. On the basis of short-term accelerated tests commercial-purity titanium, a Ti - 0.2 % Pd alloy and Hastelloy C4 have been selected as candidate metals for corrosion-resistant containers. Carbon steel and cast iron have been selected as candidates for corrosion allowance containers.

Attention is also being directed to the possibility of providing some corrosion protection for the containers. One approach would be to select a backfilling material which would inhibit attack by retarding the transport of reactants and products or by producing relatively alkaline conditions.

Another approach would be to have a multiple-layer container system. For example, a corrosive container could be placed over a corrosion-resistant container, so that if the outer vessel was penetrated locally, the corrosion of the remaining metal would cathodically protect the inner container.

In the Federal Republic of Germany the following concept of canisters are being studied for conditioned spent fuel /47/:

- metal
- graphite
- ceramic

6.3 Backfill materials, general

Backfill material occupies space between waste forms and the host rock (or liner material). It can also fill access shafts and tunnels in a mined repository. The backfill material will limit water intrusion into the repository and act as a diffusion barrier to radionuclide migration.

A great number of materials have been considered as backfill materials. Based on presently available test data and possible functions in conceptual repository designs, the following backfill materials have been identified for further study in the US: sodium bentonite, calcium bentonite, illite, quartz sand, basalt, tuff, serpentine, desiccants (MgO and CaO) and $CaSO_4$ /42/.

Fe(II)-containing mineral can be included as a redox buffer to reduce the higher valence states of the actinides and technetium and to lower the corrosion potential of the waste canister material /48/. Apatite, monazite, cinnabar, galena, chalcocite and barite can also be included to strongly sorb the actinides, iodine and strontium.

In Canada a buffer with a large fraction of smectite clay, smectite-illite clay or illite is currently favoured as a stable retardant barrier surrounding the waste package /49, 50/.

Some of the proposed backfills will contain a relatively low clay content and a large fraction of inert filler such as sand or crushed host rock. Borehole seals will be composed of compacted clay or cementation plugs, or both. Shaft seals will probably be compacted clay plugs, with grouting of the rock around these plugs if necessary. Several different seals will be employed and the volume between them will be backfilled /49/.

6.4 Repository designs

The generally accepted concept for a repository involves sinking shafts into the selected rocks and mechanical excavation of all required openings. An alternative concept is to drill an array of deep holes into the host rock from the surface.

6.4.1 The United States

The Basalt Waste Isolation Project has suggested a rather complicated waste package for emplacement in a repository at the Hanford site /51/. The details of the waste package conceptual design for spent fuel are shown in Figure 3 and 4.

Figure 3 shows the placement of waste in the canister/overpack assemblage. Pins from seven BWR disassembled fuel bundles or three PWR disassembled fuel bundles are placed in a closely packed array inside the carbon steel canister. A layer of graphite (buffer) surrounds the canister. The canister and graphite are placed inside a titanium overpack. The canister and overpack have ellipsoidally dished bottoms to minimized damage from accidentally dropping the containers during handling. The overpacked canister assemblage will be transported from the surface and emplaced in a borehole drilled in the floor of a deep tunnel. The emplacement configuration is shown in Figure 5.

Figure 4 also shows the arrangement of barrier and retrieval components external to the overpacked canister. Retrievability of the waste during the period preceding repository decommissioning is provided by a removable radiation shielding plug consisting of an aluminum container filled with a mixture of zircon sand or magnetite and bentonite. The bentonite is added to retard water penetration.

The backfill as described above is placed around the overpack. In this concept, the majority of the backfill is performed in an aluminum shell to facilitate emplacement. The aluminum can is held in place with bentonite placed between the can and the borehole.

The complete closure of the backfill layer, a backfill section is attached to the plug. This portion of backfill is enclosed in perforated sheet aluminum or aluminum mesh to protect the backfill during emplacement.

The perforations allow the backfill to expand to seal the closure against water penetration. Aluminum is used as a temporary structural metal rather then steel because the aluminum corrosion products remain in place preventing void formation, and aluminum will not appreciably affect the pH of the groundwater. The use of calcium-containing materials such as concrete is avoided to prevent exchanging the sodium in the bentonite with calcium ions, which would increase the water permeability of the bentonite.

The emplacement of spent fuel package into storage position is shown in Figure 5. The storage holes are drilled from the level floor of the storage room. The spacing between storage holes is based upon heat transfer and rock stress analysis.

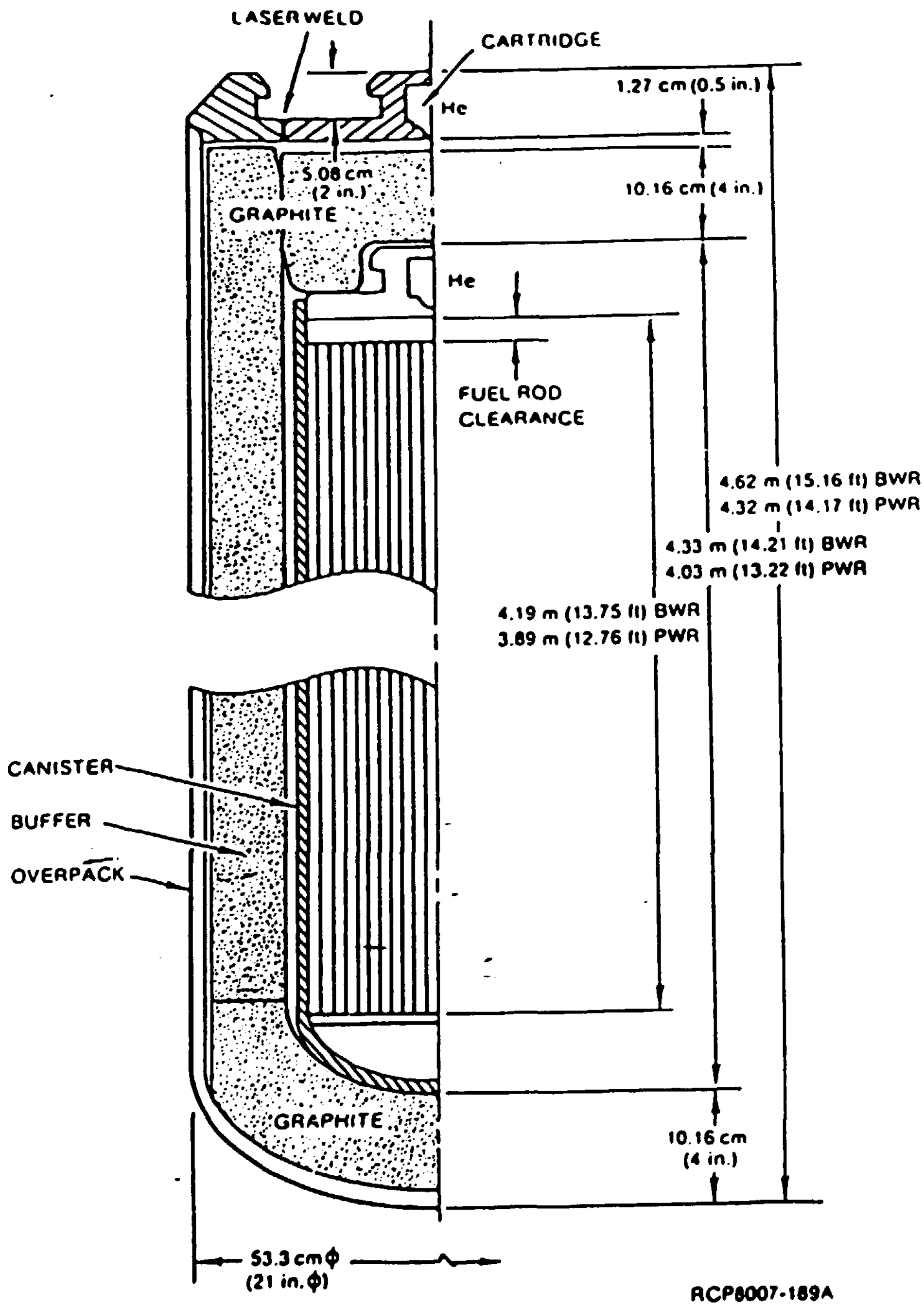

FIGURE 3. Overpacked Canister Details. /51/

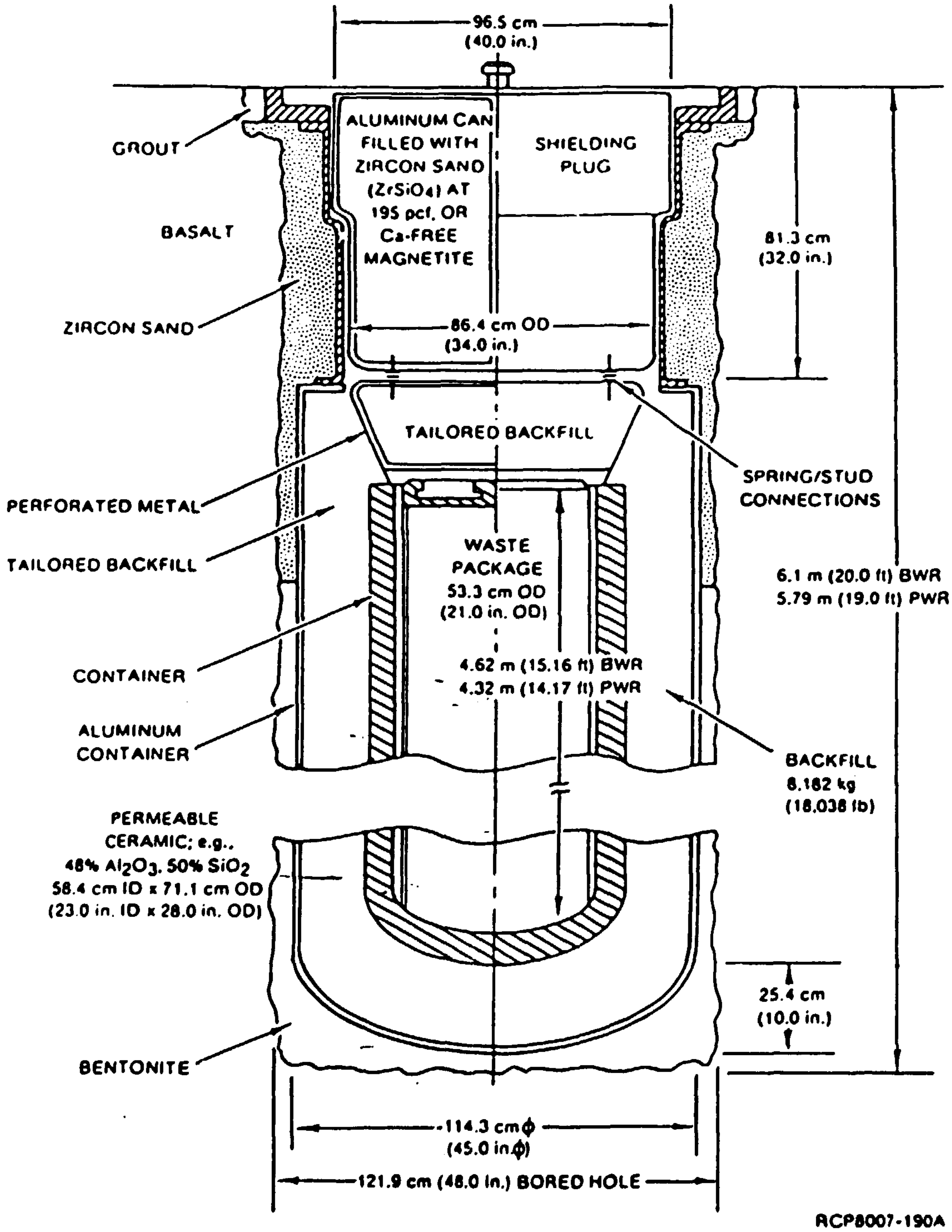

FIGURE 4. Conceptual Multiple Barrier Waste Package in Storage Hole. /51/.

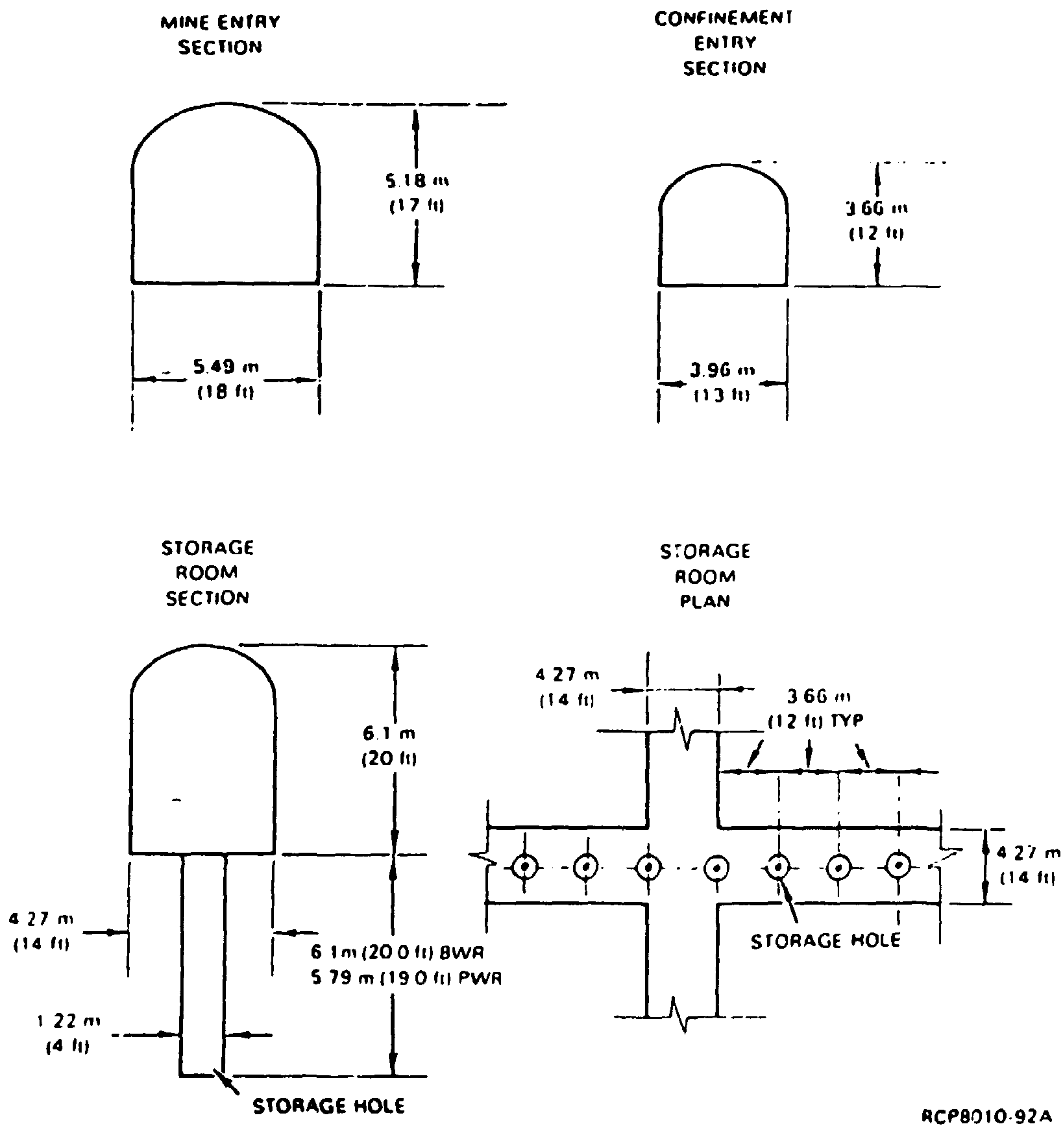

FIGURE 5. Room Cross Sections and Plan. /51/.

6.4.2 Canada

Two design concepts for a repository in igneous rock have been developed in Canada, one for spent fuel and the other for high-level reprocessing waste /52, 53/. The repositories were designed to accomodate all the high-level waste from the operation of CANDU reactors in Canada until the year 2015. The spent fuel and reprocessing waste concepts would result in the emplacement of 246 000 or 186 000 canisters, respectively, on a single level at a depth of 1 000 m. The repository would consist of a series of independent panels approx. 400 m wide.

It was proposed that spent fuel canisters, consisting of spent fuel set in lead and encapsulated in a steel container, would be emplaced on the floor of the disposal lanes in a thick layer of backfill. Backfilling would be complete after a retrievability period of 20 years. These concepts are schematically illustrated in Figure 6.

The reprocessing waste, consisting of a borosilicate glass encapsulated in stainless steel containers, would be emplaced in holes drilled in the floors of the disposal lanes, with backfilling completed immediately.

Studies of alternative vault lay-outs are in progress (54). One alternative is a disposal vault with two or more levels. Two alternative emplacement concepts will be assessed:

- Horizontal borehole emplacement: Waste packages are placed in horizontal boreholes drilled into the pillars between emplacement rooms in place of, or in addition to, emplacement in the floor of a room.

- Deep borehole emplacement: Deep boreholes are drilled in a subhorizontal array out from an emplacement room. Several waste packages are emplaced in each borehole with no retrieval option. (The thermal and buffer emplacement considerations will be of prime importance in the assessment.)

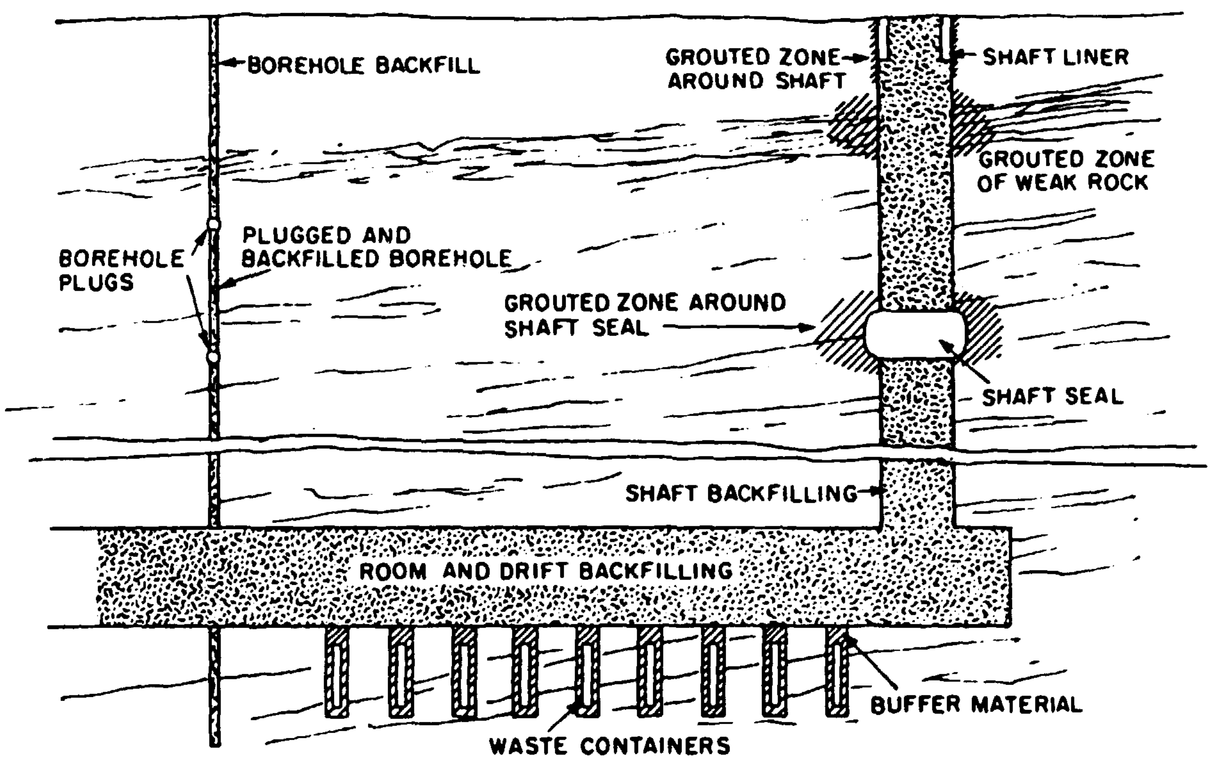

FIGURE 6. Components of Disposal Vault Sealing (Not to scale).

6.4.3 Other countries

Repository alternative in Figure 7 is presented in a Finnish report (55) and it is based upon a Swiss final disposal study (56). In the concept the canisters are placed in the middle of the tunnel in a horizontal position inside the sand/bentonite mixture. In this kind of a repository concept, the drilling of disposal holes is avoided.

In order to keep the bedrock surrounding the disposal tunnel as intact as possible, the tunnels are constructed by full-face boring which technically corresponds to the boring method of disposal holes in other alternatives. To avoid the transfer and assembling work of the boring equipment, the tunnel is spiral-like and the boring can be continuous. The emplacement of canisters can be started after one full round by using the waste transport tunnel. The technical facilities are in the middle in connection with the vertical shafts.

With regard to the construction of the repository, the most significant differences when compared with other alternatives are thus the construction of the tunnels by full-face boring and avoiding the need for boring separate disposal holes. With regard to the operation and filling of the repository, the most significant differences are the different ways of depositing the canisters and filling of the tunnels already when the canisters are emplaced.

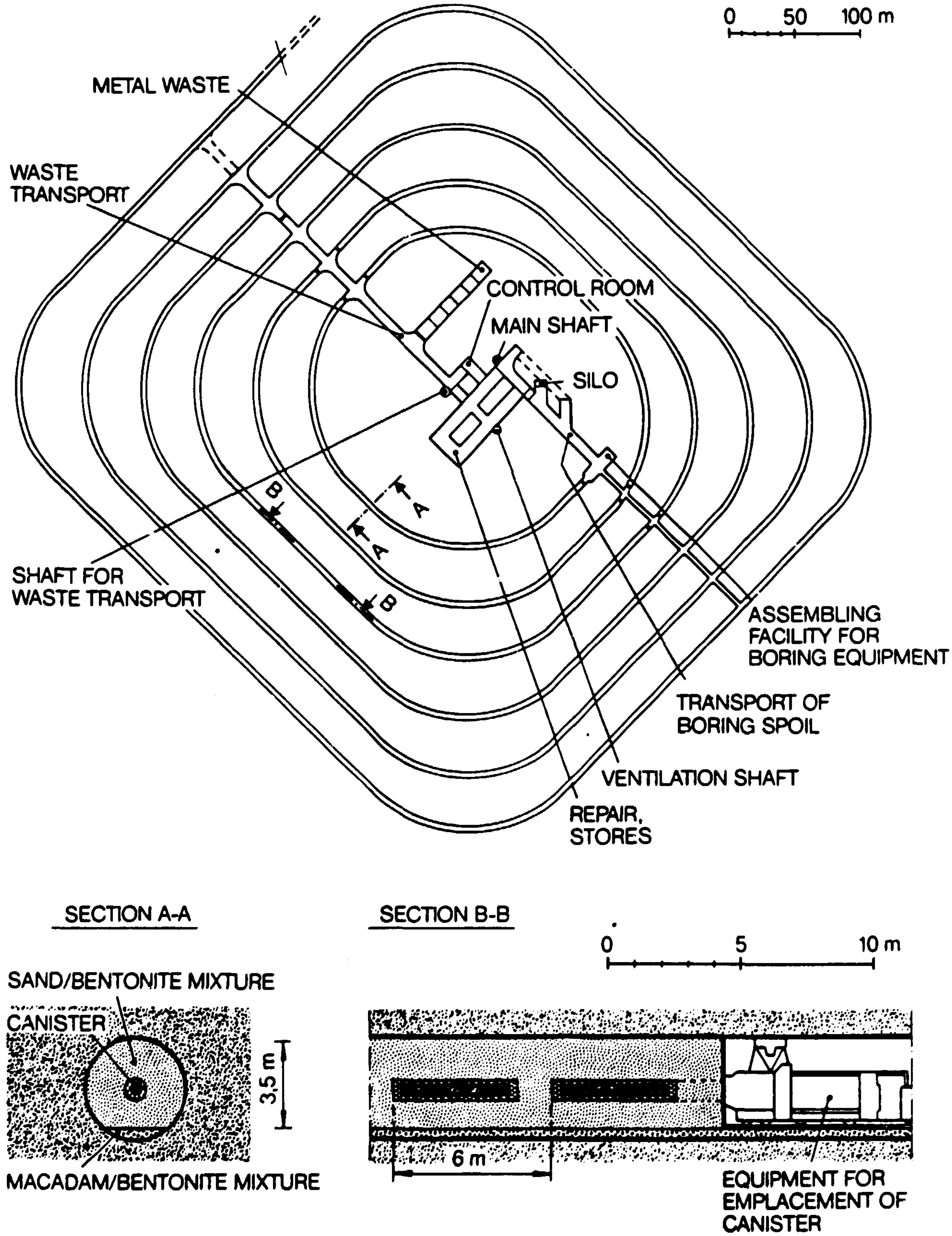

FIGURE 7. Repository alternative /55/.

7 SAFETY ASSESSMENT AND PERFORMANCE CRITERIA FOR GEOLOGIC DISPOSAL

7.1 Safety assessment

Safety assessments of proposed repository systems are required so that decisions can be made on whether to proceed with the development and operation of a repository at a particular site.

Various methods, based on deterministic and/or probabilistic approaches, can be used for safety analyses. The analysis of long-term safety is considerably more difficult to quantify than that for the operational stage, since it requires prediction of future events.

Numerous studies have been made in many countries of the potential risks to future generations from radioactive wastes in underground-disposal repositories. Some of the conclusions which can be drawn from these studies have recently been summarized in the following way /3/:

- Disruptive events that could result in direct and sudden release of waste into the environments are extremely unlikely and have consequences that are serious only within a small region near the repository site. Appropriate site selection can almost eliminate the occurrence of such events.

- The most important process for the release of waste is slow transport by groundwater. For a reasonable site location and design there are no plausible mechanisms whereby release can occur earlier than 1000 years after disposal. By this time the fission products that dominate early risk will have decayed to harmless levels. Generic risk assessment studies have demonstrated that many locations and designs are available which can ensure such containment.

- Several assessment studies have quantified the range of the potential risks to future generations from these wastes. The range in predicted risks is quite similar to those currently experienced from naturally deposited uranium ore bodies, i.e. several times natural background in some areas to many orders of magnitude below natural background in others. The pessimistic risk estimates for high-level waste disposal derive from studies that have assumed failure of engineered barriers in a poorly selected site. The very low risk estimates derive from studies that assume at least limited performance of some barriers for an average site.

- It is generally accepted that the potential level of future risk can be determined only by assessments performed for specific repository sites.

- Studies on natural elements or similar natural phenomena generally support the predictions of safety assessment studies indicating the very low migration of actinides through the geospheric pathways.

7.2 Performance criteria

The US Department of Energy published in 1981 performance criteria to be used in assessing the suitability of sites for mined geologic disposal of high-level radioactive waste /57/.

These criteria are in summary

Site Geometry	Adequate depth, thickness, and lateral extent of host medium
Host Rock Properties	Low permeability, porosity, and water content; adequate chemical, radiological, thermal, and physical and mechanical characteristics
Hydrology	Low ground water gradients; adequate distance from utilization or discharge points
Tectonic Stability	Rates and amounts of uplift or subsidence which pose no threat to the physical integrity of the repository
Faulting	Located away from active faults which would adversely affect operational safety or geological containment
Volcanism	Cannot be sited in or near an area in which igneous or volcanic activity has occurred during the Quaternary period (2 million years ago to present)

Seismicity	Must be located in regions expected to be remote from recorded or historic earthquakes of greater than "moderate" intensity
Relation to Natural Resources	Located in an area which will prevent preemptive utilization of potentially valuable resources.

In 1981 US Nuclear Regulatory Commission (NRC) issued proposed technical licensing criteria and asked for comments from federal agencies and the public /58/.

Some main points of the proposal are:

- The engineered system shall be designed so that the waste packages will contain all radionuclides for at least the first 1 000 years after permanent closure.

- The engineered system shall be designed so that, after the first 1 000 years following permanent closure, the annual release rate of any radionuclide from the engineered system into the geologic setting is at the most one part in 100 000 of the maximum amount of that radionuclide calculated to be present in the underground facility at any time after 1 000 years following permanent closure.

- The geologic repository shall be located so that pre-waste emplacement groundwater travel times through the far field to the accessible environment are at least 1 000 years.

- The geologic repository operations area shall be designed so that the entire inventory of waste could be retrieved on a reasonable schedule, starting at any time up to 50 years after waste emplacement operations are complete.

The Commission is presently analyzing the response from federal agencies and the public before issuing its final licensing criteria.

The Environmental Protection Agency (EPA) will set radionuclide release limits for the repository. Proposed rules have been issued /59/.

Some of the important points are:

Disposal systems for high-level or transuranic wastes shall be designed to provide a reasonable expectation that for 10 000 years after disposal:

a) Reasonably foreseeable releases of waste to the accessible environment are projected to be less than the quantities calculated according to Table 3.

b) Very unlikely releases of waste to the accessible environment are projected to be less than ten times the quantitites calculated according to Table 3.

Table 3. RELEASE LIMITS FOR CONTAINMENT REQUIREMENTS
(Cumulative Releases to the Accessible Environment for 10 000 Years After Disposal)

Radionuclide	Release Limit (curies per 1000 MTHM)
Americium-241	10
Americium-243	4
Carbon-14	200
Cesium-135	2000
Cesium-137	500
Neptunium-237	20
Plutonium-238	400
Plutonium-239	100
Plutonium-240	100
Plutonium-242	100
Radium-226	3
Strontium-90	80
Technetium-99	10000
Tin-126	80
Any other alpha-emitting radionuclide	10
Any other radionuclide which does not emit alpha particles	500

"Accessible environment" includes:
1) the atmosphere, 2) land surfaces, 3) surface waters, 4) oceans, and 5) parts of the lithosphere that are more than ten kilometers in any direction from the original location of any of the radioactive wastes in a disposal system.

"Reasonably foreseeable releases" means releases of radioactive wastes to the accessible environment that are estimated to have more than one chance in 100 of occurring within 10 000 years.

"Very unlikely releases" means releases of radioactive wastes to the accessible environment that are estimated to have between one chance in 100 and one chance in 10 000 of occurring within 10 000 years.

Using generalized models, EPA has assessed the long-term risk from a repository containing the wastes from 100 000 MTHM.

It is estimated that this quantity of waste, when disposed of in accordance with the proposed standards, could cause no more than 1000 premature deaths from cancer in the first 10 000 years after disposal; an average of less than one permature death every 10 years.

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