HIGH-LEVEL RADIOACTIVE WASTE DISPOSAL -

THE INTERNATIONAL SCENE

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"Radioactive Waste Management: A Geoscientific Assessment"
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ABSTRACT

The status of developments in spent fuel reprocessing, high-level waste solidification and geologic isolation is reviewed. Some generic studies on the possible range of annual radiological doses to individuals from waste repositories are discussed and compared with doses from existing nuclear power and fuel cycle operations, and with the dose received annually from the naturally occurring radiation background. The significance of neptunium-237 in dose estimation is discussed, together with technological trends in separation and disposal of long lived actinide elements.

National attitudes towards spent fuel disposal are listed in Table 1. Of 24 countries with operational nuclear power stations above 30 MW, six are currently reprocessing domestic fuel, and seven have contracts with international reprocessing services. Five countries are considering geologic burial of spent fuel. The status of national programs for fuel reprocessing is summarised in Table 2.

3. NATURE OF HIGH-LEVEL WASTE

Reprocessing of the fuel discharged yearly from a 1250 MWe LWR gives rise to about 20 m^3 of high-level waste concentrate. This is a solution of nitric acid which contains about 0.5 per cent of the uranium and plutonium, together with about 99 per cent of other actinide elements and fission products other than rare gases and iodine originally present in the spent fuel. The fission products consist of isotopes of 42 elements representing every group of the Periodic Table, from lithium (Atomic No. = 3) to ytterbium (Atomic No. = 70); 33 of these elements are present as radioactive isotopes. The actinide and transuranic elements present range from actinium (Atomic No. = 89) to californium (Atomic No. = 98). The solution spontaneously generates heat at a rate between 1 and 10 kW m^{-3}, depending on the storage time of the fuel before reprocessing.

In Belgium, France, India, Japan, the UK and the USA, high-level waste solutions from commercial nuclear fuel reprocessing have been stored in stainless steel tanks for periods up to 25 years. The tanks require cooling to remove fission product heat, secondary containment to restrain any leakages from entering the environment, redundant spare tanks for transfer of solution in the event of tank failure, and constant supervision. Storage of high-level waste solution is an interim measure before solidification of the waste into a stable, practically insoluble glass or ceramic form suitable for disposal, e.g. by burial in deep geologic formations.

4. SOLIDIFICATION OF HIGH-LEVEL WASTE

Solidification of high-level waste solution involves evaporation of nitric acid and water, calcination of residual nitrates of fission products and actinides into oxides, and conversion of the oxides into a stable waste form. Candidate waste forms include variants of glass, ceramics, and also multi-barrier matrix waste forms in which the high-level wastes are encapsulated in metal, glass or ceramic matrices. Several engineering alternatives have been developed for the evaporation and calcination operations, and for fusing, casting or pressing the waste forms. These operations must be carried out and maintained remotely owing to the high level of radioactivity.

The optimum solid high-level waste form should include the following characteristics:

- the ability to incorporate more than 52 fission products and actinide elements at a high proportion in the solidified product;
- radiation stability;
- thermal stability;
- low leachability and high resistance to groundwater attack;
- production at low temperatures to retain volatile elements; and

- production technology compatible with remote operation under highly radioactive conditions.

The ideal waste form with all the desirable characteristics has yet to be demonstrated; however, many solid forms are likely to be satisfactory in an appropriately engineered disposal system. Conversion of wastes into borosilicate glasses (vitrification) is in operation on an industrial scale for wastes from low burn-up fuel. The technology has been demonstrated on an engineering scale for wastes from high burn-up fuel; solidification of high-level waste from the annual reprocessing of fuel from a 1250 MWe LWR could give rise to about 29 cylinders of borosilicate glass, each cylinder being 30 cm diameter x 3 m height (INFCE, 1980). Glass is considered to be adequate for a first demonstration system of solidification, transportation and ultimate disposal (USNRC, 1979), but second generation ceramic waste forms are under development, e.g. SYNROC, which may have superior overall characteristics.

5. STATUS OF HIGH-LEVEL WASTE VITRIFICATION

Table 3 summarises some national achievements, developments, and plans for the solidification of high-level waste. At present most nations with nuclear power programs have selected borosilicate glass as the principal form for immobilising their high-level waste. In terms of established, engineering scale processes, borosilicate glass is the only choice available.

France is the world leader in applied vitrification technology. Following small-scale research between 1958 and 1963, a continuous industrial scale vitrification plant, Atelier Vitrification Marcoule (AVM), commenced operation in 1978 on high-level waste from reprocessing of gas-graphite reactor (GGR) fuel. The AVM plant has demonstrated an annual capacity for the vitrification of 132 m^3 of fission production solution into 60 t of glass (Chotin et al., 1982). By May 1982, 546 m^3 of high-level liquid containing 3.3 EBq (88 MCi) of radioactive waste had been vitrified in 734 canisters, each holding about 340 kg of glass. This high-level waste was produced by the reprocessing of 9657 t U of GGR fuel (Celeri et al. 1983).

The waste canisters are stored at Marcoule in concrete cells. Fission product decay heat is removed by forced air cooling; it is intended to cool the cells by natural convection after several years of storage. Maximum temperatures of the glass product are maintained below 600°C to avoid the spontaneous crystallisation of the glass, a process known as devitrification. A similar plant, Atelier Vitrification Hague (AVH), is being designed to vitrify high-level wastes from 800 t U y^{-1} of reprocessed LWR fuel at the Cap La Hague site. This capacity is equivalent to the generation of about 26 GWe of nuclear power.

By comparison, developments in waste solidification during the past 20 years in Canada, India, the FRG, the UK, the USA and the USSR have been on a laboratory or engineering pilot-plant scale. French vitrification technology is being marketed internationally; the UK has abandoned its FINGAL/HARVEST process and announced that AVM technology will be used under licence in a vitrification plant to be constructed at Windscale. Belgium and the FRG are reported to have negotiated contracts for access to AVM technology for possible application at the former Eurochemic reprocessing site at Mol and at the Karlsruhe reprocessing plant, respectively (CEA-Cogema, 1981). However, US technology for the vitrification of commercial high-level waste (Bonner et al., 1980) is believed to have been chosen for wastes from the closed US West Valley commercial reprocessing plant (Nucleonics Week, 1983).

Industrial commitment to the French vitrification technology is illustrated in Figure 1. Reprocessing commitments by France and the UK to the mid-1990s include contracts for reprocessing 9150 t U from spent LWR fuel from other countries, and 10 850 t U from domestic fuel (CEA-Cogema,1981; Jones and Pearce, 1981). High-level wastes from reprocessing 20 000 t U, equivalent to about 660 GWe y of installed nuclear power, will therefore be immobilised in borosilicate glass; this is equivalent to vitrifying wastes from more than four years of the present installed nuclear power program for the whole world (173 GWe in 1983).

6. DEVELOPMENT OF ADVANCED WASTE FORMS - SYNROC

Mineral-like crystalline ceramics have been under development at Pennsylvania State University for 10 years. These are refractory waste forms in which the fission products and actinide elements are immobilised in the crystal lattices (McCarthy and Roy, 1981). Unlike glasses, crystalline waste forms are thermodynamically stable, although they are more susceptible to radiation damage. The work is aimed at the development of a form with outstanding chemical durability which alone could minimise releases of radionuclides, even under the worst conceivable case of repository failure.

The SYNROC formulation, proposed by Professor A.E. Ringwood of the Australian National University (ANU), is a leading alternative waste form which is being developed in a collaborative program between the ANU and the AAEC. Ringwood (1980a) identified three refractory and leach-resistant minerals occurring in nature - zirconolite ($CaZrTi_2O_7$), perovskite ($CaTiO_3$) and hollandite ($BaAl_2Ti_6O_{16}$) - which have demonstrated stability for millions of years in a wide range of geologic and geochemical environments. When combined, these minerals are capable of accepting most of the high-level radioactive waste constituents into their crystal lattices.

Leach rates of alkali (Cs) and alkaline-earth (Sr) elements from non-radioactive SYNROC at 100°C into water which was replaced daily have been up to 1000 times lower than those from borosilicate glass; under static conditions, where water was saturated with leach material at 100°C, leach rates from SYNROC were typically one tenth of those from glass. The leach rate of SYNROC does not appear to increase greatly with temperature (Reeve et al., 1981). Specimens of non-radioactive SYNROC have been irradiated in a nuclear reactor to simulate radiation damage effects in waste stored up to 8×10^5 years. All specimens were intact after irradiation.

The relative merits and disadvantages of the alternative forms for immobilising commercial high-level wastes have been extensively reviewed (Johnston and Palmer, 1982; Mendel et al., 1981; CEC, 1981; IAEA, 1980; USDOE, 1981; Wald et al., 1980; USNRC, 1979). SYNROC has been ranked high in research potential, particularly for salt-bearing US defence wastes (USDOE, 1981). However, it was recently reported that defence wastes at the Savannah River Plant are to be immobilised in borosilicate glass (USDOE, 1983a) on grounds of process simplicity and cost, although similar or lower costs for ceramic waste forms had been indicated earlier (Grantham et al., 1983; Rozsa and Campbell, 1982). SYNROC could be an alternative technology for immobolising wastes from new commercial reprocessing plants in the US if the latter operation is recommenced; it has not yet attracted major interest in Europe owing to the industrial lead established by vitrification technology.

The program at the Lucas Heights Research Laboratories involves the fabrication of non-radioactive SYNROC on a multi-kilogram scale and continuation of radiation damage and leach tests (Reeve et al., 1983). This includes:

- preparation of, and leach tests on SYNROC samples containing radioactive fission products and actinides - laboratory scale equipment for manufacture of these samples is under construction;

- assessments of the engineering practicability and costs of SYNROC manufacture under highly radioactive conditions; and

- investigation of the reproducibility of the SYNROC formulation during large-scale operation.

7. STABILITY OF HIGH-LEVEL WASTE FORMS TO RADIATION

The high-level waste form will be subjected internally to intense β- and γ-irradiation from fission products, and to α- and neutron-irradiation by some actinide elements. Potential radiation damage effects include displacement of constituent atoms, rupture of chemical bonds, valency changes, transmutation of radioactive nuclei into different elements, build-up of internal energy, and deposition of helium atoms in the waste form.

The majority of radiation damage studies have been made on glasses containing high-level waste; comparatively little work has been done on the radiation stability of ceramics. Radiation studies on different waste forms have been reviewed by Mendel et al. (1981).

The effects of radiation are being studied in accelerated tests. High concentrations of short-lived β- and α-emitters will deliver in a few years doses to samples equivalent to those to be received by a real waste form over periods exceeding hundreds of thousands of years. Some tests in France and the FRG have delivered doses of β-radiation to glasses equivalent to doses from the storage of LWR wastes over several millions of years, without a detectable change in the glass structure (Amaury, 1979). In British tests, doses up to 5×10^{18} alpha decays per gram, equivalent to more than 10^5 years of storage, have been delivered to glasses; the resultant increase in leach rate was less than a factor of two (Marples, 1982; Roberts, 1980). Tests in the US have shown that changes to leachability or mechanical strength detected in glasses after α-doses equivalent to over 10^5 years of storage (Platt and McElroy, 1978) were not significant in an appropriately engineered disposal system.

8. STATUS OF GEOLOGIC DISPOSAL OF HIGH-LEVEL WASTE

Table 3 summarises national investigations of geologic formations and programs for the establishment of demonstration and commercial high-level waste repositories. Geologic media under study include bedded and domed salt deposits, crystalline rocks, basalt, clay and tuff (OECD/NEA, 1982a).

The advantages and disadvantages of the various rock types being considered for waste repositories are set out in Table 4. No formation is without some disadvantages; the IAEA (1982a) recently concluded that a site with ideal characteristics is not essential for siting an appropriately engineered waste repository.

Major design efforts have been concentrated on the placement of high-level waste in land-based mined repositories. Disposal of waste in deep drilled holes, which has been advocated by Ringwood (1980b), could be regarded as a variant of the mined repository concept. The deep-hole proposal has to some extent been neglected, although countries with small nuclear power programs have shown interest (Elsam & Elkraft, 1981).

Geologic disposal of high-level waste awaits identification of technically suitable and publicly acceptable sites; nevertheless, public opposition to the disposal of high-level waste in mined repositories has been strong. Investigations of potential geologic sites have been deferred in a number of countries and discontinued in the UK. Sweden, the FRG, Canada and the USA are leading the search with in-situ studies of their geologic formations (OECD-NEA, 1983; USDOE, 1983b; Shemilt, 1982; Salander et al., 1980).

Confidence in the disposal of high-level radioactive waste on or in the deep ocean floor as a technically acceptable alternative to land-based repositories has been expressed by France, Japan and the US (Anderson, 1982), and is the subject of considerable research effort from members of the OECD's Nuclear Energy Agency (OECD-NEA). Active participants in the OECD-NEA Seabed Working Group Program include Canada, the Commission of European Communities (CEC), France, the FRG, Japan, the Netherlands, Switzerland, the UK and the USA, and countries using observer status are Belgium, and Italy. At present, such disposal is expressly prohibited under an international convention, and major political questions on the use of the seabed are expected to occupy international forums for some years before the proposal becomes a realistic option.

9. ISOLATION OF BURIED WASTE

Driving forces which might return some of the buried radionuclides to the biosphere have been reviewed by Burkholder (1979) and are illustrated in Fig. 2. Contact of buried waste with groundwater and the leaching of radioactive constituents from the waste form is generally considered to be the most probable pathway. The release of radionuclides from waste forms by leaching involves complex mechanisms of selective leaching and matrix dissolution. These mechanisms are not completely understood.

No waste form yet developed is totally insoluble in water; the rate of leaching depends on the type and composition of the waste form (glass or ceramic), the temperature, flowrate and composition of the water, and the exposed surface area of the waste. The various radioactive elements within the waste also have widely differing leach rates. Actinide elements leach 100 times more slowly than alkali metals from borosilicate glass; also, leach rates in water at 100°C are 35-50 times greater than at 20°C (Mendel et al., 1981; Amaury, 1979).

Leach rates as low as 5×10^{-7} g cm^{-2} d^{-1} have been measured for caesium and strontium from some French glasses at 20°C. A conservative value of 10^{-5} g cm^{-2} d^{-1} has been assumed in environmental impact studies in the UK (Hill, 1979) and the USA (USDOE, 1980). This corresponds to a bulk dissolution rate of 0.014 mm y^{-1}, or a dissolution time of about 3500 years for waste fragmented into pieces 10 cm in diameter.

Potentially, the greatest leach rates are during the initial few hundred years after burial, owing to the local increase in the temperature of the geologic medium and contained groundwater through fission product decay heat. Glasses in contact with water at high temperature (300°C) and high pressure (30 MPa) undergo rapid hydration, leading to fragmentation of the sample and leaching of about 30% of the caesium from the glass in two weeks (McCarthy et al., 1978).

High temperature leach tests under reflux in the laboratory have been criticised as being unrepresentative of waste disposal conditions which could reasonably be expected in geologic burial (Savage and Chapman, 1981). At groundwater flowrates of 1 to 2 x 10^{-3} m y^{-1}, which are expected for crystalline rocks selected for waste disposal, chemical saturation effects could theoretically limit leach rates to less than 10^{-9} g cm^{-2} d^{-1} (Chapman et al., 1980). In an 88 day test under near-stagnant conditions at 200°C and 50 MPa, leach rates of borosilicate glass fell from 3 x 10^{-4} to 2 x 10^{-5} g cm^{-2} d^{-1} (Savage, 1981).

Temperatures can be lowered by reducing the concentration of fission products in the waste, storing the waste for some tens of years before burial to reduce the heat from fission product decay, and allocating adequate space for waste canisters in the geologic medium; Sweden has expressed an intention to restrict temperatures of the buried waste to below 100°C by these methods (KBS, 1978). Maximum temperatures above 350°C are apparently still under consideration for vitrified commercial high-level waste in the USA (Westinghouse, 1983). More measurements of hydrothermal leaching data are required under realistic repository conditions using actual radioactive high-level wastes and a range of geologic media (Johnston and Palmer, 1982).

However, the possibility of entry of an aquifer into a repository after major ground movement, though remote, cannot be totally excluded, and geologic burial strategy accordingly employs sequential independent barriers to retain the radioactive elements underground. Waste packages have been designed to prevent access of water to the primary waste form for up to 1000 years. In these packages, the radioactive glass or ceramic is sheathed in a corrosion resistant metal such as lead and titanium (KBS 1978), and surrounded by an absorbent overpack, e.g. bentonite clay. The overpack has two functions: to hydrate, swell and seal the inner package against major water ingress, and to provide an environment which can absorb radioactive materials which might eventually be leached from the waste form.

Isolation from the biosphere is further assured if escaping radionuclides or heavy metals are trapped by chemical interaction or adsorption processes in the repository (Roy 1981). This could occur either on the host rock or in adjacent geologic strata. Groundwater systems are determined by many complex factors. Hydrogeologic conditions at each prospective disposal site must be sufficiently understood that a reliable prediction of the possible interaction between the groundwater, the geologic medium and the emplaced waste can be made.

The complex geochemical mechanisms influencing the sorption of labile radionuclides (OECD-NEA, 1982b) indicate that engineered clay barriers may be particularly important in retention of long-lived actinide elements. Minimum performance criteria have been published for high-level waste packages and underground facilities in the US (USNRC, 1983). These are to be both designed and located so that, assuming anticipated processes and events, and including full or partial saturation of the geologic medium by water,

- the waste packages will contain substantially all radionuclides for at least 300 years, and up to 1000 years after permanent closure;

- at any subsequent time, the annual release rate of radionuclides into the geologic medium will not exceed 10^{-5} per year of the quantity remaining after 1000 years after closure, excluding those radionuclides contributing less than 0.1 per cent of the total release rate limit;

- geologic sites will be selected in areas where groundwater will take at least 1000 years to travel from the repository to the accessible environment.

10. POTENTIAL RADIATION DOSES FROM A HIGH-LEVEL WASTE REPOSITORY

Generic assessments of radiological doses resulting from ingress of water into a high-level waste burial site have been made by modelling techniques. These have involved hypothetical models for the release of radionuclides and their rates of transport to and uptake in the human environment. The models have ranged from relatively simple consequence analyses, which assumed worst case or realistic values for parameters used in the models, to sophisticated probabilistic analyses, which recognised the statistical variations in parameter values and their interactions on radiological dose (IAEA, 1981). The results of several dose assessment studies were reviewed and compared by Koplik et al. (1982).

The Swedish study (KBS, 1978) considered high-level wastes resulting from 330 GWe y of nuclear power, generated by 13 LWRs over 20 years. A total of 9000 t U of spent fuel was assumed to be reprocessed and the high-level wastes vitrified. The vitrified waste was to be stored for 40 years to reduce heat from fission product decay. Canisters of waste were to be encapsulated in lead and titanium, and buried in bentonite at a depth of 500 m in a granite formation at temperatures not exceeding 80°C.

Corrosion studies indicated that groundwater would contact the vitrified waste 1000-6000 years after burial, when temperatures had fallen to 25°C. The waste was assumed to have been fractured through earlier thermal stresses and its surface area increased fivefold. Leach rates of borosilicate waste glass, measured in the laboratory at 25°C, indicated a conservative period of 30 000 years for total dissolution of the residual radionuclides. The contaminated groundwater was assumed to percolate slowly upwards through the granite rock both to a lake and to the Baltic Sea.

Retardation and decay of the radionuclides during transport were calculated from adsorption coefficients measured on Swedish granites. Radiological doses were estimated for hypothetical populations exposed by drinking water from deep wells near the repository, using water from the lake to irrigate food crops, and by ingestion of fish and exposure to sea water andcoastal sediments. Individual radiological whole body equivalent doses derived from radionuclides from the buried waste rose to a maximum annual value of 0.13 mSv (13 mrem) after about 200 000 years. About 60% of this maximum dose was delivered by the actinide neptunium-237. This dose was equivalent to less than 7% of the annual dose from an 'average' natural background level of 2 mSv y^{-1} (200 mrem y^{-1}).

The precision of radiological dose estimates in these generic studies depends on the validity of the mathematical model and the values of parameters used for calculation. Estimates of dose from unit ingestion of certain α radionuclides have been revised on the basis of recent metabolic and radiological data (ICRP, 1979). These revisions reduced the dose estimates for radium-226 by a factor of 25, increased estimates for plutonium-239, americium-241 and curium-244 by factors of 7, 30 and 25 respectively, whereas the dose estimates for neptunium-237 were increased 250-fold. Neptunium-237 accordingly became the major potential source of dose from high-level waste repositories. Use of this revised data has resulted in significant increases in doses estimated in earlier studies (IAEA, 1982b). This is reflected in the difference between the estimate of 5 mSv y^{-1} (500 mrem y^{-1}) by Hill and Lawson (1980) and the Swedish estimate (KBS, 1978) of 150 μSv y^{-1} (15 mrem y^{-1}). An even greater maximum individual dose of 60 mSv y^{-1} (6 rem y^{-1}) was estimated from neptunium-237 in an earlier study by Hill (1979), who used conservative assumptions in the models, recognised as leading to overestimates of potential doses.

Neptunium-237 has become significant in dose estimation largely as a result of a 100 fold increase by the International Commission on Radiological Protection (ICRP) of the fractional absorption value for all neptunium compounds from the gastro-intestinal tract; the recommended value is 0.01. There is considerable uncertainty about the data, which were based on a small number of experiments on rats; absorption of trace quantities of neptunium may be a factor of ten lower, as may also be the absorption of neptunium incorporated in food (ICRP, 1979).

The high fractional absorption value may have been due to the presence of neptunium as the NpO_2^+ ion (IAEA, 1982b), and this may not be the principal chemical form of neptunium released to the biosphere from high-level waste. Evaluation of factors influencing the transfer fraction for neptunium is part of the AAEC research program at the Lucas Heights Research Laboratories.

Maximum individual radiological doses from high-level waste disposal estimated in other independent generic studies have been normalised and adjusted to the recent ICRP data and compared in Fig. 3. The consequences of repository failure in the first year after filling and decommissioning have been assessed (USDOE, 1979); however, this case was discounted as a credible situation and it was considered that detection and remedial action would protect individuals from the high estimated doses (USDOE, 1980). Other studies have assumed 1000 years of isolation for the waste before its dispersion by groundwater.

Dissolution times of the waste in groundwater, ranging from a few hundred to several tens of thousands of years, did not have a major effect on the maximum estimated doses. The effect of different food chains and the dilution mechanism indicated some advantages for siting a repository on the coast; doses were between one-sixtieth and one-five hundredth of those estimated for an inland site (Hill and Lawson, 1980). Lowest doses were estimated for waste packages disposed of on the deep ocean floor (Camplin et al., 1980), although this is expressly prohibited by an international convention.

Table 5 lists the maximum radiological doses to individuals and the major radionuclides responsible for the dose from the waste repository from some of these consequence studies. At the time corresponding to the maximum individual dose (10^4 to 2×10^5 y), neptunium-237 was predominant, contributing more than 99 per cent of the maximum. It is of interest that the contribution from plutonium-239, a radionuclide of major public concern, was estimated to be less than 10^{-6} per cent of the maximum dose.

Substantially lower maximum individual radiological doses are predicted by probabilistic scenario analyses. The Canadian Systems Variability Analysis Code (SYVAC) has been developed to model the processes occurring in the repository, the geosphere and the biosphere, and to take into account uncertainties in the data associated with those processes (Dixon and Rosinger 1981).

In SYVAC, the data obtained from laboratory and field studies are used in the form of parameter distributions rather than 'best estimate' or 'conservative' single values. A value for each parameter is sampled in turn from its distribution to form a set. This set of values defines a 'scenario'. SYVAC then determines the transport of radionuclides from the vault to the biosphere for this scenario, and estimates an individual dose. Repeated sampling produces different scenarios for which consequences are determined. Of 1730 estimates of the radiation dose to the most exposed individual, the great majority were reported to be less than 10 μSv y^{-1}, and in no case did the dose exceed 1 mSv y^{-1}.

11. COST OF HIGH-LEVEL WASTE DISPOSAL

The US Department of Energy has published (USDOE, 1979, 1980) cost estimates for the complete disposal scenario (i.e. reprocessing, vitrification, short-term storage, long-term storage and repository costs). These studies assumed manufacture of glass with a high waste loading (30 wt% calcine) and burial of waste from spent fuel 6.5 to 10 years after its discharge from the power station. These parameters resulted in estimates of temperatures of over 350°C for the buried glass. It is speculative whether such high temperatures are acceptable.

By contrast, many European countries proposed to manufacture glass at reduced waste loadings (9 - 15 wt% calcine), and to store it in near-surface facilities for several decades to reduce the heat output of the waste through fission product decay. The waste, encapsulated in packages designed to reduce water ingress for at least 1000 years, would finally be placed in geologic formations at spacings sufficient to ensure that the temperature of the external surface of the glass does not exceed 100°C. Comparative cost estimates of the US and European strategies are listed in Table 6.

The European proposals have resulted in estimates of increased costs for the vitrification operation owing to the increased volume of waste glass per tonne of spent fuel. A substantial cost penalty was estimated for extended storage of glass before geologic burial. This penalty ($US29 - 60/kg HM) (HM = heavy metal (U + Pu)) was based on appication of US-derived charges for the storage and shielding of individual waste canisters. By contrast, recent unpublished French estimates for temporary storage of vitrified glass over 30-50 years were significantly lower ($14/kg HM), but involved an unspecified degree of discounting of future costs.

In the European case, greater operating charges for geologic burial were estimated for the greater volume of waste. However, a repository of a given size could accept a greater quantity of aged waste, where both mechanical and thermal considerations were limiting factors.

12. RADIATION DOSES AND RISKS FROM A HIGH-LEVEL WASTE REPOSITORY IN PERSPECTIVE

The annual doses and risks from living near a high-level waste repository in the far distant future may be placed in perspective by comparison with doses received today from nuclear power and its fuel cycle and from the natural radiation background.

An average global annual radiation dose of 0.6 μSv y^{-1} has been estimated from nuclear power production. In the UK, the estimated average annual dose from nuclear energy is 3 μSv; maximum individual annual doses to members of the public during the late 1970s were 1.5 mSv from reprocessing, 300 μSv from nuclear reactor operation and 55 μSv from maximum upgrading and fuel manufacture (NRPB, 1981). British Nuclear Fuels Ltd estimated a lower maximum annual dose of 880 μSv for reprocessing in 1981 (BNFL, 1982).

Estimates of average annual radiological doses from natural background have recently been increased to 2 mSv y^{-1} (UNSCEAR, 1981). Thorium in beach sands at Kerala, India, contribute to an abnormally high background of up to 20 mSv y^{-1}. In Australia, sunbathing for 2 hours per day on Kingscliffe Beach in New South Wales can result in an external dose of 320 μSv y^{-1}, which is equivalent to a continual exposure rate of 3.8 mSv y^{-1}.

The presence of uranium in igneous rocks results in increased natural background doses from the accumulation of radon gas and its decay products in dwellings. A recent survey in granite areas of the south-west and northern UK has indicated doses of 5 mSv y^{-1} in 100 000 houses, 25 mSv y^{-1} in about 1000 houses, with maximum doses of 100 mSv y^{-1} (NRPB, 1983). Regulatory proposals include a design level of 5 mSv y^{-1} for new buildings and a rehabilitation action level of 25 mSv y^{-1} for existing houses.

Living in a brick dwelling rather than a wooden home can increase the annual individual dose by about 0.7 mSv y^{-1}. A return visit by air to the UK from Australia incurs an individual dose of about 0.3 mSv. By comparison, the maximum individual dose estimated in consequence analyses for waste repository ranged from 0.13 mSv y^{-1} (KBS, 1978) to 60 mSv y^{-1} (Hill, 1979). Probabilistic analyses suggested doses less than 1 mSv y^{-1} (Dixon and Rosinger, 1981).

The risk of death from radiation-induced cancer averaged over age and sex (NHMRC, 1977) is about 1.25×10^{-2} Sv^{-1}. Exposure to a natural radiation background of 2 mSv y^{-1} carries, therefore, a proportionate death risk of 2.5 in 100 000 per year of exposure.

A similar statistical level of risk applies to the following societal activities in the UK (Flowers 1976):

- smoking 40 cigarettes,
- travelling 2000 km by car,
- travelling 10,000 km by plane,
- rock climbing for 40 minutes,
- canoeing for 2.5 hours,
- engaging in ordinary factory work for about 40 weeks, and
- simply being a human aged 60 for 8 hours.

The upper limit of individual radiological dose estimated for the Swedish repository (KBS, 1978) and by Hill (1979) corresponded to about one-eighth and 30 times the risk from these activities respectively; the SYVAC analysis suggested one-half of these risks. These levels of individual risk must be weighed against the benefit of the 330 GWe y of electrical energy produced.

13. TECHNOLOGICAL TRENDS

An alternative form of high-level waste disposal which has been studied intensively by the CEC involves separation of the long-lived actinides from the fission products. The actinides would be recycled through nuclear reactors and transmuted into nuclides with shorter half-life and reduced radio-toxicity. Fission products and transmuted actinides would be buried in deep geologic formations. While partition and transmutation of actinides seems technically feasible, it has been considered unlikely to be cost effective in dose reduction (IAEA, 1982b).

Actinide separation was however strongly supported by the Working Group on Spent Fuel Management, chaired by Professor Castaing, in a recent report to the French Supreme Council for Nuclear Safety (CSSN, 1982). The Castaing Report recommended the development of processes for the removal of the majority of neptunium-237 from high-level waste, together with americium-241 and curium-245, which form neptunium-237 by radioactive decay. The separated actinides would undergo special treatment, including immobilization in ceramics such as SYNROC, or nuclear transmutation. Industrial implementation of these processes in France was envisaged by the end of the century. The Group further recommended against geologic disposal of unreprocessed spent fuel or vitrified high-level waste at the present time.

The relative quantities of neptunium-237 contained in high-level wastes from generation of 1 GWe y of electrical power in an LWR and a Liquid Metal Fast Breeder Reactor (LMFBR) are illustrated in Table 7. The data refer to a period following 5000 years of geologic burial, when the americium and curium precursors will have substantially decayed into neptunium (CSSN, 1982). The table shows that about 2.7 times more neptunium-237 would be present in unreprocessed spent LWR fuel than in high-level wastes from standard reprocessing operations, the difference resulting from the unrecovered plutonium-241 in the unreprocessed fuel. Vitrified high-level waste from reprocessing LMFBR fuel could contain about 60% of the neptunium-237 from reprocessing LWR fuel for the same power generation, reflecting the different fission characteristics of the reactor types. Removal of 90% of neptunium-237 is more effective in reducing its inventory in geologic burial of LWR high-level wastes than in the case of LMFBR waste; to achieve the greatest potential reduction in the neptunium inventory requires the removal of the precursors americium and curium before burial.

The case for actinide separation has not been accepted internationally. Initial reactions are that it could increase capital and unit costs of reprocessing by 20 - 25% (Nuclear Fuel, 1983). Future trends seem likely to be concentrated on resolving the uncertainties in the metabolic and radiobiological data for neptunium and on improving mathematical modelling techniques using realistic hydrologic and geochemical input data.

14. CONCLUSIONS

(a) Many solid waste forms appear suitable for disposal in appropriately engineered repositories without presenting a significant biological hazard.

(b) Borosilicate glass continues to be the principal radioactive waste form selected by many nations and has been used on an industrial scale since 1978.

(c) SYNROC is generally accepted as an improved radioactive waste form offering great development potential.

(d) The technology for the disposal of radioactive wastes in some geologic formations is available.

(e) The procedures for geologic disposal require, among other things, input from advanced hydrogeologic organisations for the development of regional hydrogeologic models for the prediction of groundwater movement.

(f) More assessment is required of the consequences of repository failure during the first few hundred years after waste burial.

(g) Further study is necessary to improve confidence in the validity of and sensitivity to parameter values in mathematical models used to estimate radiological dose resulting from failure conditions, particularly with reference to neptunium-237.

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TABLE 1

ATTITUDES TOWARDS REPROCESSING IN COUNTRIES WITH NUCLEAR POWER STATIONS ABOVE 30 MW

OPTIONS FOR DISPOSAL OF SPENT FUEL

GEOLOGIC BURIAL OF UNREPROCESSED SPENT FUEL BEING CONSIDERED BY:

- CANADA
- FINLAND
- SWEDEN
- SWITZERLAND
- USA

REPROCESSING AND BURIAL OF SOLIDIFIED HIGH-LEVEL WASTE

COUNTRIES REPROCESSING:

- FRANCE
- INDIA
- JAPAN
- UK
- USSR
- FEDERAL REPUBLIC OF GERMANY (FRG) **

COUNTRIES CONTRACTED FOR REPROCESSING SERVICES:

- BELGIUM
- ITALY
- JAPAN
- NETHERLANDS
- SWEDEN
- SWITZERLAND
- FEDERAL REPUBLIC OF GERMANY (FRG)

UNCOMMITTED:

- ARGENTINA ***
- BULGARIA *
- CZECHOSLOVAKIA *
- FINLAND *
- GERMAN DEMOCRATIC REPUBLIC (GDR) *
- HUNGARY *
- KOREA
- PAKISTAN ***
- SPAIN
- TAIWAN
- YUGOSLAVIA

* Spent fuel of Soviet origin ultimately to be returned to USSR.

** Plant understood to be under maintenance.

*** Pilot scale reprocessing plants reported under construction.

TABLE 2

NATIONAL PROGRAMS FOR REPROCESSING SPENT FUEL FROM COMMERCIAL NUCLEAR POWER GENERATION

COUNTRY	REPROCESSING CAPACITY AND PLANS
Belgium	The 60 t U y^{-1} Eurochemic plant at Mol was operated between 1966 and 1974; it was then closed as uneconomic. A decision on recommencement of operations and possible increase in capacity is awaited. Belgium has contracted for French reprocessing of about 54 t U of fuel.
Canada	Research supporting vitrification development.
German Democratic Republic (GDR)	Spent fuel is to be returned to the USSR for reprocessing.
France	Natural uranium gas-graphite fuel has been reprocessed since 1958 in the 1200 t U y^{-1} UP1 plant at Marcoule, and also since 1967 in the 900 t U y^{-1} plant at Cap La Hague. The UP2 plant, after adaptation, commenced reprocessing LWR fuel in 1976 at a nominal capacity of 100 t U y^{-1}. This capacity is being progressively expanded, and new plant, UP2-800, with 800 t U y^{-1} capacity is scheduled for operation by 1989. A third plant, UP3A, of capacity 800 t U y^{-1} is under construction and expected to operate in 1987. A duplicate plant, UP3B, is also under consideration. France has international reprocessing contracts involving a total of about 6000 t U of LWR fuel.
Finland	Spent fuel of Soviet origin is to be returned to the USSR for reprocessing.
Federal Republic of Germany (FRG)	The experimental 35 t U y^{-1} WAK plant at Karlsruhe, operational since 1971, was reported closed in May 1980 for repairs to equipment. Construction of a 350 t U y^{-1} plant has been proposed at Dragahm in Lower Saxony and at Wackersdorf in Bavaria. A total of 1700 t U of spent LWR fuel is contracted for reprocessing in France.
India	The 60 t U y^{-1} Trombay plant, reprocessing natural uranium metal fuel, became operational in 1965. The 100 t U y^{-1} Tarapur plant for reprocessing HWR and LWR fuel became operational in 1977. It is believed that a third plant of capacity 100 t U y^{-1} for spent oxide fuel from heavy water reactors is intended to be operational in the late 1980s at Kalpakkam.

TABLE 2 (Cont'd.)

COUNTRY	REPROCESSING CAPACITY AND PLANS
UK	The 1000 t U y^{-1} B204 plant reprocessed natural uranium gas-graphite fuel in the 1950s and early 1960s. The 2000 t U y^{-1} B205 plant has reprocessed this fuel since 1964 and is scheduled for a major renovation. The B204 plant, after modification, reprocessed LWR fuels between 1968 and 1973. A 1200 t U y^{-1} thermal oxide reprocessing plant (THORP) is under construction and expected to commence LWR fuel reprocessing by 1990. The UK has international contracts for reprocessing about 3100 t U of spent fuel.
USA	The 300 t U y^{-1} plant at West Valley, N.Y., operated intermittently from 1966 to closure in 1972. A novel 300 t U y^{-1} plant at Morris, Illinois was never operated owing to maintenance design problems. Reprocessing of commercial nuclear power fuel was deferred indefinitely in 1977. Construction was halted on the 1500 t U y^{-1} plant at Barnwell, South Carolina which could require an additional $800 M to complete. Its operation in the 1990s has been suggested.
USSR	Spent fuel reprocessing is being carried out on a pilot scale; no data are available on the capacities or locations of Soviet reprocessing plants. However it is understood that spent fuel of USSR origin arising in COMECON countries, e.g. Bulgaria, Czechoslovakia, GDR, is scheduled for return to the Soviet Union. The USSR has also negotiated for return of similar spent fuel from Finland.
Italy	20 t of LWR fuel has been contracted for reprocessing in the UK. The Eurex pilot plant at Saluggia has capacity equivalent to about 10 - 20 t U y^{-1} of LWR fuel and is used for reprocessing research and development.
Japan	A small 100-140 t U y^{-1} demonstration reprocessing plant at Tokai Mura has been in operation intermittently since 1977 on LWR fuel. A commercial 1200 t U y^{-1} plant for reprocessing LWR fuel is planned for operation in 1990. Japan has contracted for reprocessing 1600 t U LWR fuel in France and 1600 t U in the UK. Japan has also renewed a contract for reprocessing 500 t U of gas-graphite fuel in the UK.
Netherlands	120 t U has been contracted for reprocessing in France.
Sweden	727 and 140 t U of LWR fuel have been contracted for reprocessing in France and the UK respectively. The majority of Swedish spent fuel (6000 t) is to be stored in Sweden for up to 20 years in a central store (CLAB) pending a decision on its reprocessing or disposal.
Switzerland	470 t U has been contracted for reprocessing in France.

TABLE 3

STATUS OF SOLIDIFICATION AND GEOLOGIC BURIAL OF HIGH-LEVEL WASTES FROM COMMERCIAL THERMAL NUCLEAR POWER GENERATION

Country	High-level Waste Solidification		Geologic Burial	
	Preferred Waste Forms	Developments	Preferred Formations	Developments
Australia	SYNROC	Non-radioactive large-scale production development, together with leaching trials at high temperatures and pressure on inactive and irradiated specimens is being conducted. Tests with specimens containing fission products and actinides are planned.	--	--
Belgium	Borosilicate glass beads in lead matrix (PAMELA Vitro-met), borosilicate glass blocks.	A vitrification plant (PAMELA) using FRG technology is scheduled for operation at Mol by 1986, and a second plant using French technology is scheduled for 1987.	Clay	An underground laboratory under construction at Mol is planned for operation in 1982 to conduct in-situ tests in the clay deposit relevant to geologic disposal.
Canada	Borosilicate and aluminosilicate glasses, glass beads/lead alloy, matrix, crystalline ceramics, glass-ceramics and unreprocessed fuel.	Nepheline syenite glass blocks containing aged fission products have been subjected to a long-term leaching trial since the early 1960s. The Waste Immobilization Process Experiment (WIPE) currently under construction, is a demonstration vitrification unit with a production capacity of 10 kg h^{-1} of glass.	Granite	Construction of an underground laboratory at Whiteshell is planned to study geology, mining techniques, heat transfer and back filling performance. Generic research in the 1980s is planned to be followed by a demonstration repository in the 1990s, and by a commercial repository after 2000 for disposal of either reprocessed fuel or HLW.
Denmark			Salt	Salt domes in Jutland are under examination. The program includes field studies, design and safety assessments for a repository supporting a 6000 MWe nuclear program for 30 y, with burial assumed from 2010.
German Democratic Republic (GDR)			Salt	A waste repository is being developed at Bartensleben.
Finland			Crystalline Rock	Risk assessment studies, economic evaluations.

TABLE 3 (Cont'd.)

Country	High-level Waste Solidification		Geologic Burial	
	Preferred Waste Forms	Developments	Preferred Formations	Developments
France	Borosilicate glass	The industrial scale AVM plant at Marcoule commenced operation in 1978. About 540 cubic metres of HLW corresponding to reprocessing of over 9000 t of natural uranium gas-graphite fuel were vitrified into 250 t of glass by early 1981. A larger plant of similar design, AVH, is scheduled to be operational at Cap La Hague in 1986 to solidify waste from up to 1600 t U y^{-1} LWR fuel, corresponding to a nuclear generation capacity of about 50 000 MWe.	Granite	It is intended to store glass blocks for 30-50 years in air cooled vaults. Subsequent burial at 1 km depth is being considered.
India	Borosilicate glass	Commissioning of a vitrification plant at Tarapur commenced in 1982. The plant has a capacity of about 120 kg d^{-1} of glass. Similar plants are planned at Kalpakkam and at Bhabha Atomic Research Centre.	Granite	Burial after 30 y of storage in an air-cooled underground vault is planned.
Italy	Borosilicate glass	Small scale demonstration vitrification plants are under consideration.	Clay, Salt	The sediments near the Trisaia Centre in S. Italy are under study.
Japan	Borosilicate glass, zeolite ceramics.	Vitrification technology has been developed since 1976. The Vitrification Pilot Plant is planned for construction in the late 1980s.	Granite, Diabase, Shale, Zeolitic Tuff, Limestone, Schist	Storage of glass for up to 50 years. Underground disposal in Japan on an experimental basis may commence in 2015.
Netherlands			Salt	Inland salt domes are under study; former consideration of salt domes under the North Sea has been discontinued on grounds of cost.
Sweden	Borosilicate glass; unreprocessed fuel.	------	Granite	A 3-year international study of geologic disposal commenced in 1981 at Stripa and is an in-situ experiment scheduled for completion in 1986.
Switzerland	Borosilicate glass	------	Granite	Feasibility studies for burial at depths of up to 1500 m are in progress.

TABLE 3 (Cont'd.)

Country	High-level Waste Solidification: Preferred Waste Forms	High-level Waste Solidification: Developments	Geologic Burial: Preferred Formations	Geologic Burial: Developments
Spain	Borosilicate glass	Use of the West German PAMELA process under licence is under consideration.	Salt, Shale, Ceramite	A pilot plant repository is under consideration for the late 1980s.
UK	Borosilicate glass	The FINGAL vitrification process was developed between 1962 and 1966. This technology was the basis for the HARVEST engineering scale work commenced in the early 1970s. This national program was abandoned in 1981. French technology is being used for construction of an industrial scale vitrification plant scheduled to be operational in the late 1980s.	Granite	Some experimental drilling tests have been conducted in N. Scotland but were discontinued in December 1981. It is proposed to store the vitrified waste at the surface for at least 50 years.
USA	Unreprocessed fuel; borosilicate glass, titanate ceramics, SYNROC, cermets, cement, coated particles, super calcine, metal matrix materials.	The waste solidification engineering prototype plant (WSEP) had a capacity solidifying wastes from 1 to U d^{-1} of fuel. Nearly 2 EBq (50 MCi) of radioactive waste was encapsulated between 1966-1970. The Nuclear Waste Vitrification Plant (NWVP) in 1979 vitrified high-level waste containing about 0.4 MCi from reprocessing 1.5 t U of LWR fuel. 260 kg of glass was produced in spray calciner in-can melters, 20 cm diameter x 2.4 m height, which served as encapsulating canisters for the waste.	Salt, Granite, Basalt	A waste isolation pilot plant (WIPP) in New Mexico is under construction in salt for operation by late 1980s for disposal of transuranic waste of military origin. The Nevada terminal waste isolation project in granite is in use as a test site for temporary (5 year) geologic emplacement of commercial spent fuel. The first National Terminal Waste Storate (NWTS) site is scheduled for operation by 1998.
USSR	Aluminophosphate glasses.	No details available.	Salt	Studies have been reported on near-surface storage facilities for vitrified high-level wastes. Over 10^8 Ci has been reported disposed of by injection of liquid waste containing about 1 Ci L^{-1} into deep sandstone formations
Federal Republic of Germany (FRG)	Borosilicate glass, glass ceramics, phosphate glasses.	A vitrification plant, PAMELA, using FRG technology is under construction at Mol. Pilot Scale vitrification development is being carried out at Karlsruhe.	Salt	The disused Asse salt mine, used as a repository for low-level radioactive wastes since 1967 is being used for in-situ testing of thermal dissipation in salt. The salt dome at Gorleben is currently being investigated for possible siting of a high-level waste repository; a decision on site suitability is expected in the early 1990s.

TABLE 4
GEOLOGIC FORMATIONS FOR ULTIMATE DISPOSAL OF HIGH-LEVEL WASTES

Type of Formation	Advantages	Disadvantages	Investigating Countries
Rock salt comprising bedded salt and dome salt (from diapirs)	1. High thermal conductivity 2. Plastic condition at low temperature and pressure ensures fractures are self-sealing 3. The existence of rock salt indicates its isolation from circulating waters throughout geologic times past 4. Very low permeability	1. A natural resource commonly associated with other resources such as oil and gas 2. Subject to creep under load that accelerates with increase in pressure associated with deep burial, and increase in temperature associated with heat-generating radioactive wastes 3. May contain up to 3% bitterns in which the salt is highly soluble under the influence of heat 4. Bitterns that migrate to the waste canister walls in response to canister heat form pockets of highly corrosive fluid 5. Many radioactive elements in waste will dissolve if they come in contact with bitterns 6. Poor ion-exchange properties	German Democratic Republic (GDR) Netherlands USA Federal Republic of Germany (FRG)
Crystalline rock (Granite, gneiss, granodiorite, gabbro, etc.)	1. Underground openings are self-supporting or require minimum support 2. Fractured-rock permeability decreases with depth, and many such rocks have very low permeability 3. Weathered rock adjacent to fractures has ion-exchange properties 4. Large diameter holes can be drilled to great depths where temperatures exceed 300°C	1. Will be a major source of energy in the future, i.e. rock with a thermal gradient of 20°C per 1000 m 2. Fractures are not self-sealing 3. Low permeability back-fill material required to seal repository 4. The larger the underground opening, the more fractures intersected	Austria Canada Denmark France India Japan Sweden Switzerland

TABLE 4 (Cont'd.)

Type of Formation	Advantages	Disadvantages	Investigating Countries
	5. Low thermal gradients are present in some areas (10°C per 1000 m) 6. The regional groundwater system can be accurately modelled to predict flowlines and travel times		UK* USA
Basalt	As for crystalline rock above except for (2) and (6): 2. Zones with very low permeability do exist, but permeability does not necessarily decrease with depth 6. The regional groundwater system can be modelled, but it is more complex owing to variations in numbers of fractures between individual basalt flows and interbedded tuffs	As for crystalline rock above except for (1), and the addition of another point: 5. Because some basalts are highly fractured and permeable, selection of a low permeability thickness of basalt will require considerably greater effort in proving its suitability than would be required for crystalline rock	USA
Argillaceous formations (1) Clay	1. Good plastic characteristics and unsupported openings would be self-sealing 2. No current value as a resource 3. Very low permeability within the clay itself 4. Good ion-exchange properties	1. Subject to creep and all openings will require full support and full lining 2. The construction of a fully lined repository at depths greater than 250 m may not be possible or the cost may be prohibitive 3. Highly permeable sands are commonly interbedded with clays 4. Chemical reactions take place within the clay with increase in temperature, and a formation temperature above 100°C is not recommended	Belgium Italy

* Note: The UK recently decided [UK Hansard, 1981] to cease current geological exploration activities in view of its intention to prolong temporary storage of high-level waste.

TABLE 4 (Cont'd.)

Type of Formation	Advantages	Disadvantages	Investigating Countries
		5. Little information is available on the migration of fluids within the clay in response to a thermal gradient	
(2) Shale and tuff	1. Less plastic than clay and unsupported openings would readily deform 2. No value as a resource unless containing hydrocarbons 3. Very low permeability expected, but some variation between individual beds 4. Good ion-exchange 5. Large diameter holes can be drilled to considerable depths and kept open by the use of dense circulating muds	1. All openings require support and full lining 2. The construction of a repository would be possible at depths greater than 250 m, but deformation of the installation would increase with depth 3. Some highly permeable beds may be part of the sequence 4. Reaction to a thermal gradient expected to produce changes in the clay minerals and their properties	Italy USA

TABLE 5

RADIOISOTOPES CONTRIBUTING TO THE MAXIMUM ANNUAL INDIVIDUAL RADIOLOGICAL DOSE FROM A HIGH-LEVEL WASTE REPOSITORY

Reference	KBS (1978)	KBS modified*	Hill (1979)	Hill & Lawson (1980)	Burton & Griffin (1981)	Time of Maximum Dose
Radionuclide	mSv y^{-1}	mSv y^{-1}	mSv y^{-1}	mSv y^{-1}	mSv y^{-1}	y
^{237}Np	0.1	25	60	5	10	$10^4 - 2 \times 10^5$
^{99}Tc	2×10^{-2}	6×10^{-2}	7×10^{-2}	5×10^{-2}	10^{-2}	$200 - 6 \times 10^3$
^{226}Ra	2×10^{-2}	2×10^{-3}	3×10^{-2}	10^{-3}	2×10^{-4}	$5 \times 10^4 - 10^6$
^{233}U	3×10^{-2}	10^{-1}	6×10^{-7}	4×10^{-7}	4×10^{-5}	$2 \times 10^4 - 5 \times 10^4$
^{135}Cs	6×10^{-3}	6×10^{-3}	3×10^{-4}	2×10^{-5}	3×10^{-4}	$9 \times 10^4 - 10^6$
^{234}U	3×10^{-3}	10^{-2}	10^{-5}	4×10^{-7}	10^{-6}	$3 \times 10^4 - 10^6$
^{129}I	6×10^{-4}	10^{-3}	8×10^{-2}	5×10^{-2}	--	$200 - 6 \times 10^3$
^{239}Pu	3×10^{-7}	2×10^{-6}	2×10^{-9}	2×10^{-11}	3×10^{-9}	$4 \times 10^5 - 3 \times 10^6$

* Data adjusted to revised dose factors in ICRP (1979).

TABLE 6

COMPARATIVE COSTS OF HIGH-LEVEL WASTE DISPOSAL (US$ 1978)

Operation	European Strategy: Glass - 10 wt% calcine				USDOE Strategy: Glass - 30 wt% calcine			
	Capital (10^6 $)	Annual Operating (10^6 $)	Economic life (y)	Levellised costs $/kg HM	Capital (10^6 $)	Annual operating (10^6 $)	Economic life (y)	Levellised costs $/kg HM
Predisposal								
Spent fuel transport & storage				~60[C]				~60[C]
Spent fuel reprocessing	700	35-59	15	197-208	700	35-59	15	197-208[C]
HLW solidification	91.4	17.9	15	20.0[C]	55	7.1	15	10.4[C]
Rail transport				3.33[C]				3.3[C]
Extended storage at solidification plant or geologic repository	127	34.0 (27.3 y)	100	29.6[F] 60.4[C]	n.a.	n.a.	n.a.	n.a.
Geologic repository	2053	65.3	25	31.95[F]	2053	33.1	16.4	38.1[F]

n.a. - Not appropriate
F - Federal funding
C - Commercial funding

HM - Heavy metal
HLW - High level waste

TABLE 7

NEPTUNIUM CONTENT OF HIGH-LEVEL WASTES AFTER 5000 y OF GEOLOGIC BURIAL PER GWe y GENERATED IN AN LWR AND AN LMFBR
(From CSSN, 1982)

REACTOR TYPE	REPROCESSING OPTION*			
	Unreprocessed Fuel	Standard Reprocessing	90% Removal of Np	90% Removal of Np, Am, Cm
	^{237}Np kg GWe^{-1} y	^{237}Np kg GWe^{-1} y	^{237}Np kg GWe^{-1} y	^{237}Np kg GWe^{-1} y
LWR	60.9	22.7	9.3	2.4
LMFBR	Not Applicable	13.1	10.5	1.6

* Reprocessing assumed 3 y and 1 y following discharge from LWR and LMFBR with 99.6% removal of Pu. Reprocessing is an essential part of the LMFBR fuel cycle.

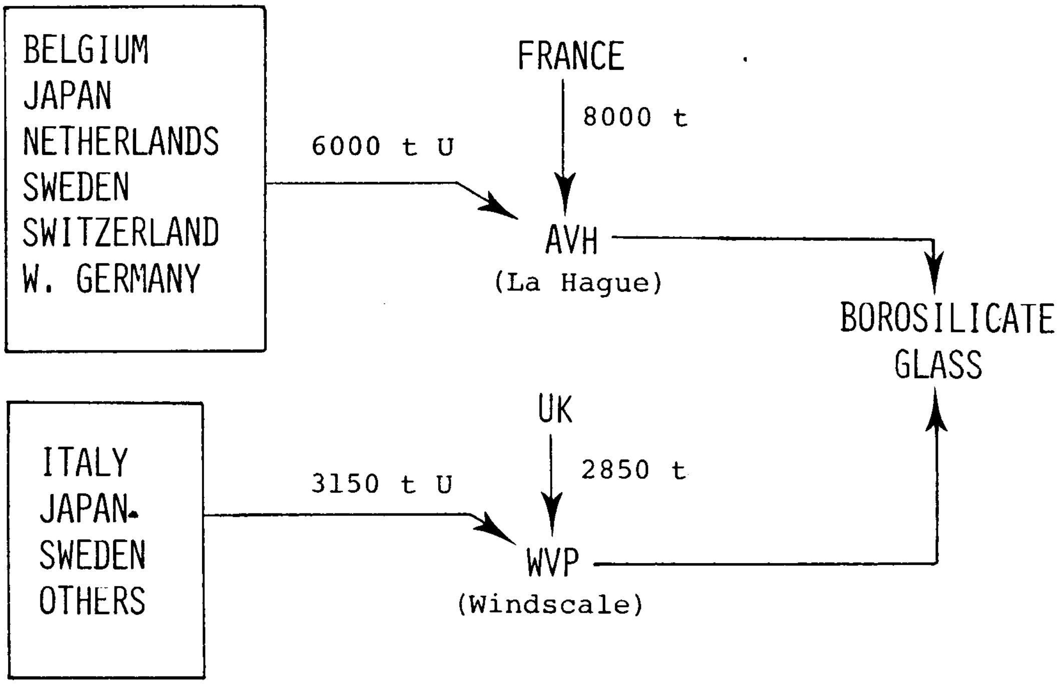

FIGURE 1. REPROCESSING COMMITMENTS BY FRANCE & UK TO 1995

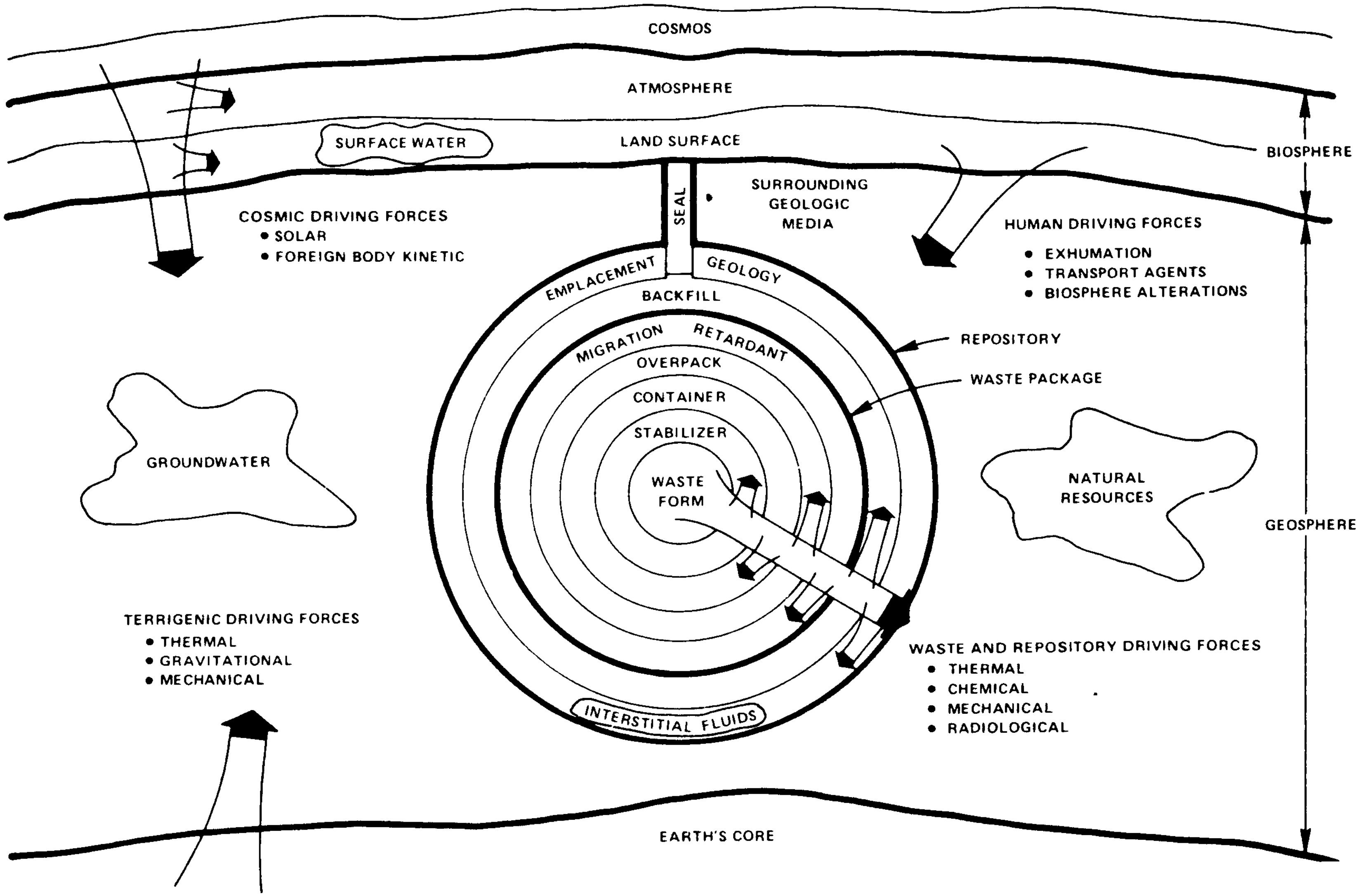

FIGURE 2. WASTE ISOLATION SYSTEM AND DRIVING FORCES (After Burkholder, 1979)

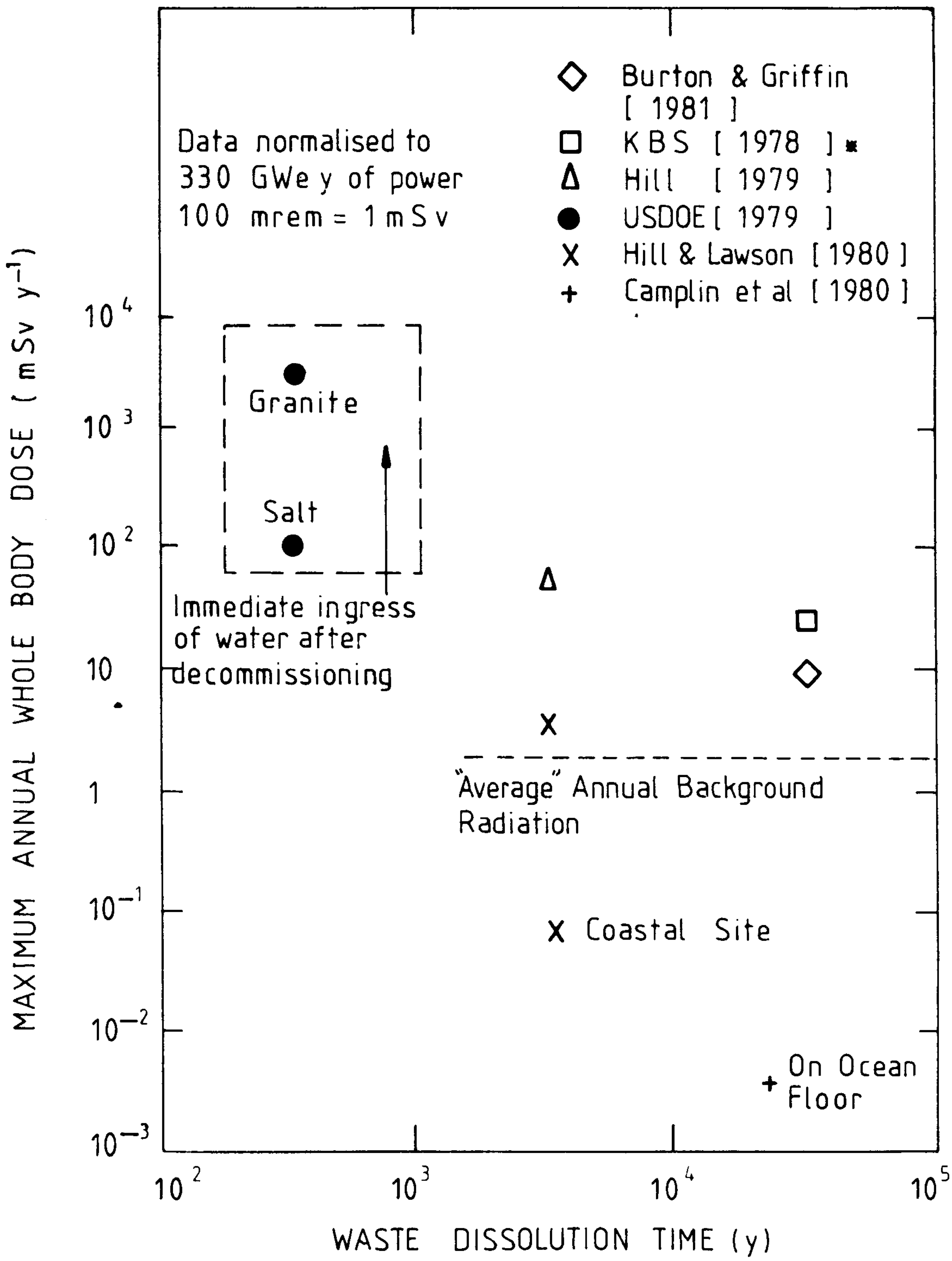

FIGURE 3. ESTIMATE OF RADIOLOGICAL DOSE FROM A HIGH-LEVEL REPOSITORY

* using dose factors from ICRP (1979)