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Characteristics and Quantities of Radioactive Wastes [and Discussion]

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Characteristics and quantities of radioactive wastes

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Estimates are given of the total quantities of radioactivity, and of the contribution from different isotopes to this total, arising in the wastes from civil nuclear power generation; the figures are normalized to 1 GW(e) y of power production. The intensity of the heat and γ-radiation emitted by the spent fuel has been calculated, and their decrease as the radioactivity decays. Reprocessing the spent fuel results in 95% or more of the fission products and higher actinides being concentrated in a small volume of high-level, heat-emitting waste. The total decay curve of unreprocessed spent fuel or of the separated high-level waste is dominated by the decay of some fission products for a few hundred years and then by the decay of some actinide isotopes for some tens of thousands of years. The residual activity is compared with that of the original uranium ore. Some of the long-lived activity will appear in other waste streams, particularly on the fuel cladding, and the volumes and activities of these wastes arising in this country are recorded. The long-lived activity arising from reactor decommissioning will be small compared with the annual arisings from the fuel cycle.

1. Introduction

The purpose of this introductory paper is to summarize the dimensions of the subject that the papers in this Discussion Meeting are to address. We give calculations and estimates of the radioactive wastes arising from typical nuclear fuel cycles, and of the total wastes produced by the nuclear industry in this country.

The quantity and radioactive characteristics of the wastes arising from civil nuclear power programmes are determined both by the size of the programme and by the details of the composition of the fuel and of the management of the spent fuel. The radioactive content of the waste products is dominated by the number of fission reactions that have taken place, which is determined in practice simply by the size of the nuclear programme, while the volume and chemical nature of the wastes depends on the decision whether or not to reprocess the spent fuel. The data presented here can therefore illustrate reasonably well the nature of radioactive waste arisings associated with any civil nuclear power programme. Wastes from research reactors can require different techniques of treatment but the total quantities are small compared with wastes from power generation and they are therefore ignored, as are arisings from other industrial or medical uses that tend to contain only short-lived species. No estimates are given in this paper of radioactive wastes arising from military programmes because these data are not usually available; as far as this country is concerned, the Radioactive Waste Management Advisory Committee (1981) has stated 'Defence wastes pose problems which are similar in kind, though smaller in scale, than those of the civil nuclear programme'.

The fission of uranium or other actinide atoms in nuclear reactors leads to the production of a large number of radioactive species, arising from: (1) the fission process in nuclear fuel;

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(2) neutron-capture reactions arising in the nuclear fuel itself; (3) the neutron activation of fuel cladding and of reactor components in a neutron flux, and the contamination of coolant circuits. The quantities and decay characteristics of the radioisotopes arising in typical nuclear fuel cycles are summarized in the next section. The complete fuel cycle will include wastes arising from uranium mining and enrichment, which contain long-lived species, though the total quantity of radioactivity is low and these wastes are dilute enough to be treated as 'low-level waste'; nevertheless, the quantities arising are noted below.

The radioactivity arising in reactor components, (3) above, tends to be short-lived and depends on the type of reactor; some examples are given in the later sections. The most important processes are those occurring in the fuel, (1) and (2) above. The actual bulk and the chemical composition of wastes contaminated with radioactive components depend on the details of the fuel handling and reprocessing routes adopted, and examples taken from current practice in this country are given in §5.

The precise inventory of fission products and actinides will vary between different fuels and in different nuclear reactors because the isotopic composition of the nuclear fuel as a function of time will depend on: (1) the isotopic composition of the original fuel; (2) the neutron spectrum in the reactor; (3) the detailed management of the fuel; and (4) the duration of irradiation. In practice, the radioactive wastes arising from fuel cycles based principally on either ^{235}U or ^{239}Pu fission are quite similar; the differences are not enough to affect the engineering problems met in their management. Most of the data in this paper therefore refer to pressurized water reactors (PWR) using slightly enriched uranium fuel, since this type of reactor is the most common in the world. For comparison, some data are given for wastes from the gas-cooled 'Magnox' reactors used in this country and in France, and for liquid-metal cooled fast breeder reactors.

To facilitate comparison with other compilations, the results quoted here are usually normalized to the production of 1 GW(e) y of electricity, that is, the output from a reactor generating 1 GW for 1 year, using stated values for the reactor efficiency. Assuming a PWR has a net efficiency of 32.5 %, the production of 1 GW(e) y would require a total thermal output of 3.1 GW, and involve approximately 3×10^{27} fission events per year. A simplified mass flow in a 'once-through' fuel cycle through a PWR generating continuously at 1 GW(e) is shown in figure 1. In this example, taken from Flowers (1983), 200 t of natural uranium is required to produce 37 t of 3 % enriched uranium in the oxide fuel; after generation of 1 GW(e), 35.6 t of uranium remains in the spent fuel, along with about 1.2 t of fission products and 0.3 t of plutonium. This mass flow is typical of a thermal reactor fuel cycle.

Perhaps the most constructive comparison is with a typical fast reactor fuel cycle. Figure 2 shows the annual mass flow calculated for a conceptual liquid metal-cooled fast breeder reactor

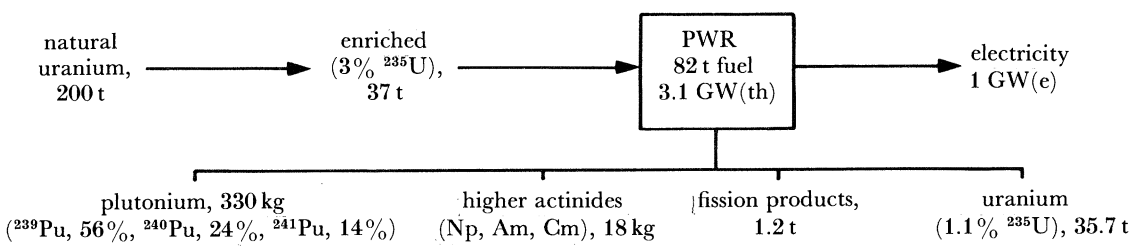

FIGURE 1. The mass flow in a 'once-through' fuel cycle in a PWR generating at 1 GW(e) for 1 year.

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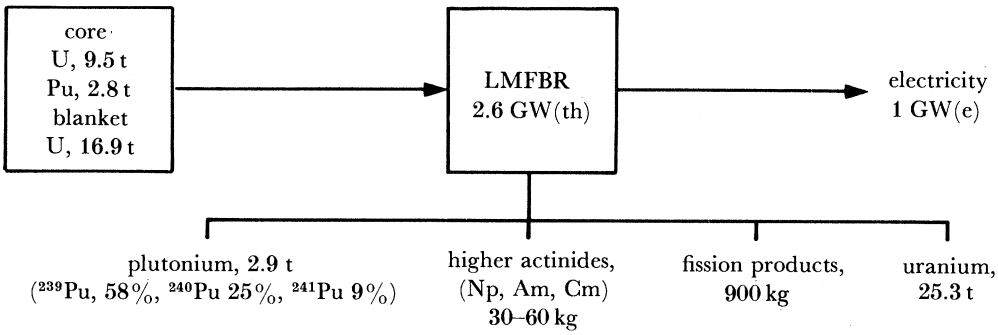

FIGURE 2. The mass flow through a liquid metal-cooled fast breeder reactor generating at 1 GW(e) for 1 year.

(LMFBR) generating continuously 1 GW(e), assuming a mixed plutonium-uranium oxide fuel in the core using plutonium from various sources, and a blanket of depleted uranium. For each 1 GW(e) of output, the total mass of fission products is somewhat less than for the PWR, because of the higher assumed reactor efficiency, while the output of higher actinides is larger. In this example, taken from Flowers and from Farmer (1983), the net gain of plutonium is 0.18 t but this quantity is, of course, very dependent on the details of reactor design. Repeated cycling of reprocessed plutonium through an LMFBR (the 'equilibrium' fuel cycle) would not yield a result very different from that shown in figure 2.

2. FISSION PRODUCT AND ACTINIDE INVENTORIES

Detailed inventories of the nuclides in irradiated material have been calculated by using the FISPIN computer code. The code is described in detail by Burstall (1979) and is supported by several validation documents. Data on nuclear reaction cross sections, fission yields and nuclear half-lives are required for FISPIN calculations. The data are assembled into libraries that can be selected by the user who will specify additional parameters such as the initial fuel composition and irradiation conditions. The currently recommended data libraries contain data on approximately 100 actinides and 800 fission products. Validation and improvement of the data are undertaken throughout the nuclear industry in the U.K. Output from the code consists of fundamental data on the quantity and activity of each nuclide and other derived quantities such as γ-ray spectra, decay heat output and spontaneous neutron emission.

The half-lives, mode of decay and activity of the most important long-lived isotopes which are contained in fuel at the time of discharge from a PWR are recorded in table 1. This example assumes that 3% enriched uranium fuel has been irradiated to 32000 MW d t^{-1}, which would mean a dwell time for fuel of about 2.2 years at full power in a 1 GW(e) reactor. The total activity at discharge is 2.2×10^8 TBq/GW(e) y, with a rapid decay (in 10 years' storage) to 4×10^5 TBq due to the decay of a wide range of radionuclides with short half-lives, from a few minutes to about a year. These include the fission products ^{89}Sr, ^{95}Zr, ^{103}Ru, ^{127}Tl, ^{129}Te, ^{131}I and ^{141}Ce, and also a number of nuclides generated by the neutron activation of structural components of the fuel elements and impurities in the fuel, e.g. ^{60}Co, ^{59}Fe, ^{54}Mn and ^{65}Ni. It has been assumed that PWR fuel elements consist of uranium oxide fuel clad in zircalloy but that the fuel assemblies include end plates and spacers of stainless steel; the total assembly consists of 82% zircalloy, 13% of stainless steels and 5% of Inconel.

The decay curve of the total radioactivity of discharged PWR fuel and that of the principal

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longer-lived nuclides in the fuel is plotted in figure 3. This figure also includes the decay curve for natural uranium fuel irradiated to 4800 MW d t^{-1} in one of the gas-cooled 'Magnox' reactors, though the data for the individual isotopes are not plotted in this case for the sake of clarity. Inspection of the figure shows that the total activity to be handled and disposed of in the spent fuel is very similar in each case and this will be so for every thermal reactor, such as the advanced gas-cooled reactors (AGR), which are now operating and being built in this country.

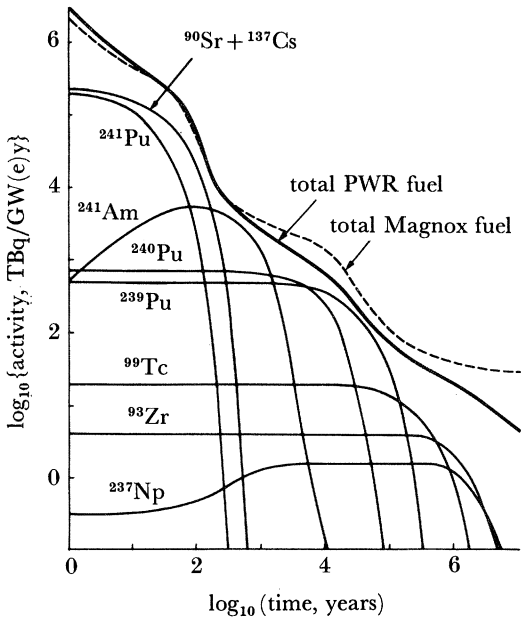

FIGURE 3. The total radioactivity of discharged PWR fuel as a function of time, normalized to the production of 1 GW(e) y. The contributions of some of the main radionuclides are indicated. The curve for the total radioactivity of discharged Magnox fuel is shown for comparison.

TABLE 1. ACTIVITY AT DISCHARGE FOR PWR UNREPROCESSED FUEL

(32000 MW d t^{-1} burn-up)

nuclide	half-life (years) and main mode of decay		activity at discharge from reactor (TBq per GW(e) y)	nuclide	half-life (years) and main mode of decay		activity at discharge from reactor (TBq per GW(e) y)
^{59}Ni	7.5×10^4	K cap	18.1	^{238}U	4.47×10^9	α	0.412
^{90}Sr	28.5	β	9.60×10^4	^{237}Np	2.14×10^6	α	0.304
^{93}Zr	1.5×10^6	β	3.87	^{239}Pu	2.44×10^4	α	438
^{99}Tc	2.1×10^5	β	18.3	^{240}Pu	6540	α	663
^{129}I	1.57×10^7	β	0.0338	^{241}Pu	14.9	β	1.70×10^5
^{135}Cs	2.0×10^6	β	0.42	^{242}Pu	3.87×10^5	α	2.65
^{137}Cs	30.1	β	1.32×10^5	^{241}Am	433	α	170
^{234}U	2.44×10^5	α	0.0129	^{243}Am	7400	α	22.5
^{235}U	7.04×10^8	α	0.0245	^{245}Cm	8350	α	0.214

One other feature which is common to all reactor types is that the total activity is dominated by the β-emitters ^{137}Cs and ^{90}Sr for the first 300 years and then by the α-emitters ^{241}Am and ^{239}Pu to about 100000 years. The principal source of ^{241}Am is the ^{241}Pu isotope, which decays

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with a half-life of 14.9 years. At longer times, the important isotopes are the β-emitters ^{99}Tc and ^{93}Zr, and the α-emitters ^{237}Np and ^{242}Pu; the ^{237}Np increases to a maximum after 1000 years because it is formed from the decay of ^{241}Am.

3. Long-lived species in the natural uranium series

The wastes arising from the extraction of uranium from ores, usually termed mill tailings, constitute a particular class of low-level waste, in which the radioactivity is naturally occurring and long-lived. The uranium is separated from the ore by physical and chemical processes such as grinding, dissolution and precipitation. The tailings are what remain, together with some of the liquors from the extraction process. Ores currently considered economic range from 0.07% to 0.4% of U_3O_8; typically the volume of tailings is about 30000 m^3 produced from an ore at the upper end of the range to supply the 200 t of uranium needed to produce 1 GW(e) y from a PWR.

After the majority of the uranium has been extracted, most of the original radioactivity, i.e. the uranium daughters from ^{230}Th downwards, remain in the mill tailings. The members of this decay chain are listed in table 2. The activity of each of the 10 members of this series in the ore needed to provide 200 t of uranium would be 2.5 TBq, assuming they were at secular equilibrium in the ore and none had been lost by geologial processes.Thus the total activity could be up to 25 TBq. In addition, there could be contributions from the decay series of ^{235}U and of ^{232}Th if there were thorium minerals associated with the uranium ore.

Table 2. Principal radionuclides in mill tailings

radionuclide	half-life	mode of decay
^{230}Th	80000 years	α, γ
^{226}Ra	1620 years	α, γ
^{222}Rn and short-lived daughters	3.8 days	α, γ, β
^{210}Pb	20.4 years	β, γ
^{210}Bi	5.0 days	β
^{210}Po	138 days	α

Despite the low specific activity of these milling wastes, some precautions have to be taken to prevent the dispersion of radioactive dusts and prevent the contamination of ground waters. It is also necessary, because of radon emissions, to prevent building taking place over a tailings site and to prevent material being removed for building elsewhere. The collective dose arising from the escape of radon from the mill tailings was estimated to be a large component of the total collective radiation dose from a 'once-through' uranium fuel cycle by the International Fuel Cycle Evaluation Exercise (I.A.E.A. 1980).

4. Spent fuel management and disposal

Spent fuel is always stored with artificial cooling on discharge from a reactor because of the heat output associated with the intense radioactivity. In a 'once-through' fuel cycle, the spent fuel would be treated as waste and directly disposed of in engineered repositories after a period of storage. Alternatively, the spent fuel can be reprocessed so that the plutonium and residual

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uranium may be reused. Fuel from a sodium-cooled fast reactor would always be reprocessed because of the high inventory of plutonium.

To illustrate the practical problems involved in spent fuel management, it is useful to calculate the heat output and γ-energy arising from the inventory of nuclides. The heat output from discharged PWR fuel elements as a function of time is plotted in figure 4, again normalized to show the heat output per GW(e)y of power output, including an allowance for the activation products present in typical cladding material. A very similar curve for Magnox spent fuel is again included for comparison. The practical benefits of storage with air or water cooling for many decades before disposal, to limit the temperature rise in a repository, have been discussed by Hodgkinson *et al.* (1983) both for spent fuel disposal and for the similar case of the disposal of vitrified high-level waste.

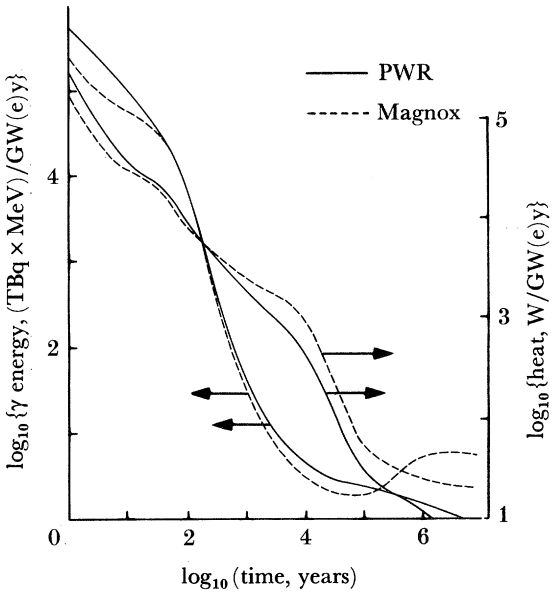

FIGURE 4. The heat output and intensity of γ-energy from discharged PWR fuel as a function of time.

The amount of radiation shielding required in storage or transport is determined by the intensity and energy of the γ-radiation. To illustrate how this quantity changes with time, a plot of the sum of the product of activity and γ-radiation energy for each nuclide (TBq × MeV) is also given in figure 4, again for the PWR and Magnox discharged fuels. The important isotopes are produced by ^{137}Cs decay during the first 300 years, and ^{241}Am thereafter for about 3500 years. The data in figure 4 are not easily converted to actual shielding requirements because these will depend on the precise geometry of any assembly, which will affect the self-shielding contribution. As one practical example, five PWR fuel elements, each containing 0.46 t of uranium, at 300 days after discharge could be shipped in a transport flask consisting of 165 mm lead and 170 mm steel, to give a γ-ray dose rate at 1 m of about 0.02 mSv h^{-1}. Very roughly, the thickness of shielding required to give the same dose rate would reduce to 37 mm lead and 170 mm steel after storage for 100 years and to 65 mm of steel after storage for 1000 years. The neutron dose arising from spontaneous fission or (α, n) reactions must also be taken into account; this can be controlled by using suitable neutron-absorbing materials.

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5. Practical consequences of reprocessing fuel

Reprocessing metal or oxide fuel involves dissolving it in nitric acid and performing several cycles of solvent extraction to yield highly purified uranium and plutonium products and a variety of waste streams dependent on the precise chemistry adopted. Uniformly, however, the product from the first extraction cycle is high-level waste, containing more than 95% of the activity; this is normally evaporated to a small volume and is concentrated enough to require artificial cooling during storage. Small fractions of the total inventory of plutonium, actinides and fission products will appear in other waste streams, all of which will eventually be incorporated in solid waste forms, apart from final washings that are of sufficiently low activity to be discharged to the environment. The Radioactive Waste Advisory Committee recommended that, in the U.K., solid waste in which heat production need not be taken into account be classified as intermediate level (ILW) if the specific activity exceeds 4 GBq t^{-1} of α activity or 12 GBq t^{-1} of β activity and as low level (LLW) if the specific activity is lower than these values but greater than 400 kBq t^{-1} (1984).

The intermediate-level waste will contain also the long-lived neutron activation products from the reactor components. These arise mainly from the fuel cladding and other components of the fuel assemblies, which are separated from the fuel material itself during the first stages of reprocessing.

In the reprocessing cycles used now, the higher actinides, Am and Cm, follow the lanthanide fission products into the high-level waste. The fate of Np is more complex, because of its variable valency in aqueous solution. ^{237}Np is formed in irradiated reactor fuel by three routes:

$$^{235}\mathrm{U}\ (\mathrm{n},\gamma)\ ^{236}\mathrm{U}\ (\mathrm{n},\gamma)\ ^{237}\mathrm{U}\ (\beta)\ ^{237}\mathrm{Np}; \quad (1)$$

$$^{238}\mathrm{U}\ (\mathrm{n},2\mathrm{n})\ ^{237}\mathrm{U}\ (\beta)\ ^{237}\mathrm{Np}; \quad (2)$$

and

$$^{241}\mathrm{Pu}\ (\beta)\ ^{241}\mathrm{Am}\ (\alpha)\ ^{237}\mathrm{Np}. \quad (3)$$

^{237}Np from routes (1) and (2) is present in the fuel when removed from the reactor and for a PWR constitutes about one quarter of the eventual total. ^{237}Np from route (3) is delayed by the 15 years half-life of ^{241}Pu and the 430 years half-life of ^{241}Am; depending upon the time that elapses before the fuel is reprocessed, the ^{241}Am and the ^{237}Np will be formed in the separated plutonium product or in the high-level reprocessing waste. Eventually, however, assuming that the plutonium product is irradiated in new fuel, almost all the ^{237}Np will be present either in spent fuel or in high-level waste. Depending upon the exact flow-sheet used, some ^{237}Np may pass into the later cycles of the reprocessing plant where it is separated in intermediate-level waste streams. The behaviour of actinides in the reprocessing of fast reactor fuel is broadly similar.

High-level waste

Further papers in this Discussion Meeting deal with the processes that can be applied to transform HLW into a solid form; in what follows, we shall assume that the HLW liquor will be solidified; the resulting solid will, of course, still require cooling, and the heat emission and γ-energy will fall in a manner similar to that illustrated for spent fuel in figure 4.

The decay curve for HLW from that quantity of reprocessed PWR fuel used in generating 1 GW(e) y is shown in figure 5, assuming that reprocessing takes place at 5 years after discharge, and that all but 0.1% of the uranium and plutonium is removed during reprocessing. The Am

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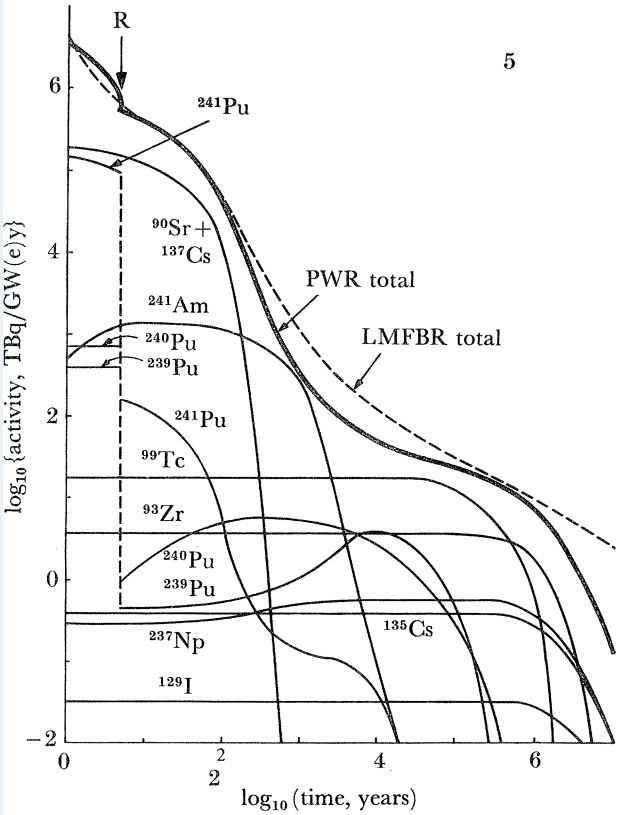
and Cm follow the fission products into the HLW; it has been assumed that all the initial ^{237}Np is also directed to the HLW. The great difference between this decay curve and that for the unreprocessed fuel in figure 3 is due to the separation of virtually all the plutonium isotopes, reducing markedly the residual activity between 300 and 10^5 years; indeed, after some 10000 years, the total activity of HLW falls below that of the quantity of ore containing the 200 t of uranium needed to generate 1 GW(e) y, which is *ca.* 34 TBq.

A similar decay curve for HLW from a fast reactor fuel is plotted in figure 6. In this example, the original plutonium feed was assumed to come from reprocessing AGR fuel (though the source makes little difference), and the fuel attained a peak fuel burn-up in the core of 10 % and was reprocessed 1 year after discharge to remove all but 0.1 % of the plutonium and 0.1 % of the uranium. For ease of comparison, the total decay curve for fuel from a fast reactor is also plotted in figure 5. It may be seen that the use of a plutonium fuel and the higher actinide content of the fast reactor fuel on discharge makes relatively little difference to the overall decay curve.



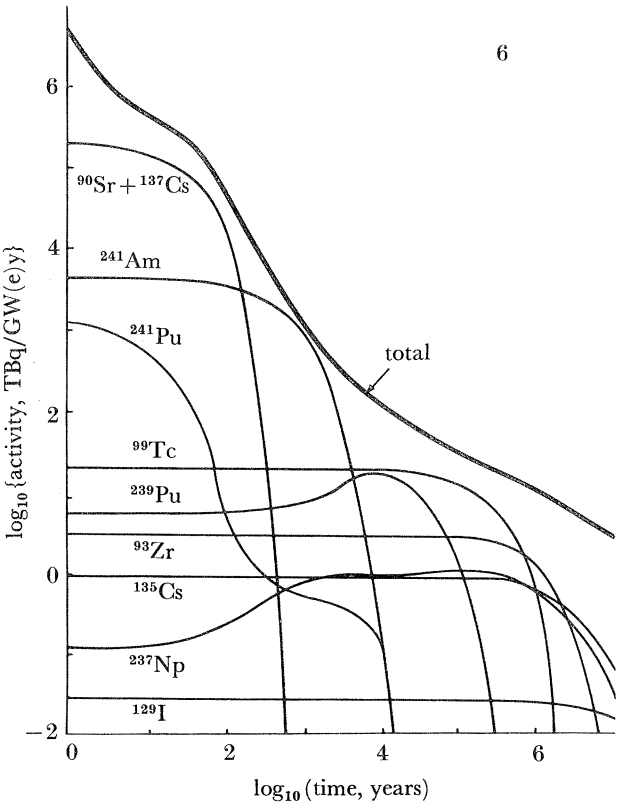


FIGURE 5. The radioactivity of high-level waste from reprocessed PWR fuel as a function of time, with the contributions from some individual radionuclides (R denotes the time of reprocessing). The decay curve for high-level waste from a liquid metal-cooled fast breeder reactor is included for comparison.

FIGURE 6. The radioactivity of high-level waste from reprocessed fuel from a liquid metal-cooled fast breeder reactor as a function of time, with the contributions from some individual radionuclides.

Intermediate-level waste streams

Intermediate level wastes containing long-lived activity occur in the waste streams of the reprocessing plants, from plutonium fuel fabrication plants and from research activities. The total activity arising in ILW from the reprocessing plants is simply the residual that is not routed

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to the HLW. The volumes of ILW are more difficult to estimate, because they depend on the precise form and waste loading of packages, which will differ for each waste stream. For long-term storage and disposal, all such wastes will be incorporated into a durable, solid form. While final decisions have not yet been taken, it is probable that most, if not all, types of ILW in the U.K. will be incorporated into blocks of concrete, and the volumes of packaged wastes have been estimated on this assumption, though the packaging processes and hence the final volumes will be subject to change as development proceeds.

To put the matter into perspective, table 3 records the approximate volumes and total activity of the packaged ILW that will arise in this country to the year 2000 and to the year 2030 on the assumptions made for a 'medium' nuclear programme as defined by Jenkins (1983) at the recent Sizewell Inquiry, i.e. about 20 GW(e) installed in 2000 and 30 GW(e) in 2030. The data in table 3 are derived from the inventory of wastes prepared by Fairclough *et al.* (1985). The essential assumption, apart from the size of the programme, is that the Magnox reactors will be shut down by 2005, and be replaced by either AGR or PWR reactors. The low fuel rating of the Magnox reactors compared with the more advanced reactor types means that the mass of fuel to be reprocessed (per GW(e)) is larger, and the volumes of ILW correspondingly larger, than arise from the more highly rated oxide fuel.

TABLE 3. CONDITIONED VOLUMES AND ACTIVITIES OF WASTES IN THE U.K. TO THE YEARS 2000 AND 2030

	conditioned volume		total activity, TBq			
	10^4 m^3		α		β, γ	
waste year...	2000	2030	2000	2030	2000	2030
high-level (vitrified)	0.13	0.31	4.5×10^5	1.4×10^6	1.2×10^8	2.0×10^8
intermediate-level:	1.1	6.3	30	46	4.3×10^3	2.3×10^5
reactor operation						
reprocessing	5.2	9.8	3.3×10^4	6.0×10^4	3.5×10^6	3.8×10^6
research and industrial	1.7	5.3	1.0×10^3	2.7×10^3	4.5×10^5	7.6×10^5

It can be seen from table 3 that the volumes of long-lived ILW containing appreciable α activity are some 40 times greater than the volumes of HLW, but the total activity (α or β) is one fortieth; the specific activity is therefore lower by a factor of around 1600. Some such ratio is liable to be typical of many reprocessing régimes, and a fast reactor fuel cycle should not give rise to a very different result, though the volumes of ILW contaminated with plutonium will tend to be larger when a plutonium fuel is used.

Most of the long-lived activity in ILW is found on the fuel cladding and is recorded under 'reprocessing' in table 3. A typical prediction for the decay of PWR cladding wastes is shown in figure 7, calculated for the same reactor régime as exemplified above and on the assumption that all but 0.2% of the adhering fuel is removed during the fuel dissolution process. The activity is dominated by the activation products of the neutron bombardment of stainless steel, and in the long term by nickel isotopes. A quite different result is obtained for Magnox cladding wastes in the U.K. because the Mg alloy cans are stripped mechanically from the fuel rods and are not chemically processed. The cladding swarf consists of Mg with some Al and about 1% of nimonic alloys. But the adhering fuel particles are not as efficiently removed as by chemical dissolution and the dominant activities are the residual plutonium isotopes and fission

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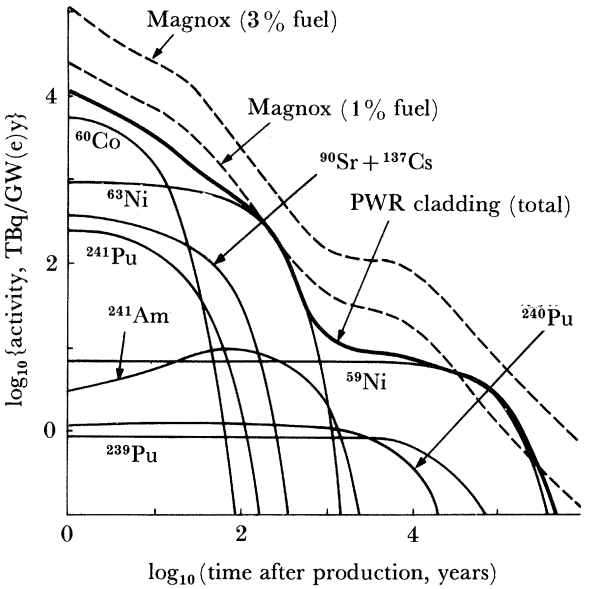

FIGURE 7. The radioactivity of PWR and Magnox cladding waste as a function of time. Some individual contributions to the activity of the PWR cladding waste are shown. The two decay curves for Magnox cladding wastes are for 3% and 1% retention of fuel on the cladding.

products. Estimates of the resulting activity averaged over the Magnox programme and normalized to 1 GW(e)y are also shown on figure 7 assuming that either 1% or 3% of the fuel is retained on the cladding. The individual isotopes are not recorded in the figure for the Magnox cladding wastes.

Reactor decommissioning wastes

Once the fuel and fuel assemblies are removed from a reactor, the remaining structure should containing little long-lived activity except the metallic activation products such as ^{59}Ni or ^{63}Ni; the specific activities will, however, be much lower than for the fuel cladding. Woollam (1981)

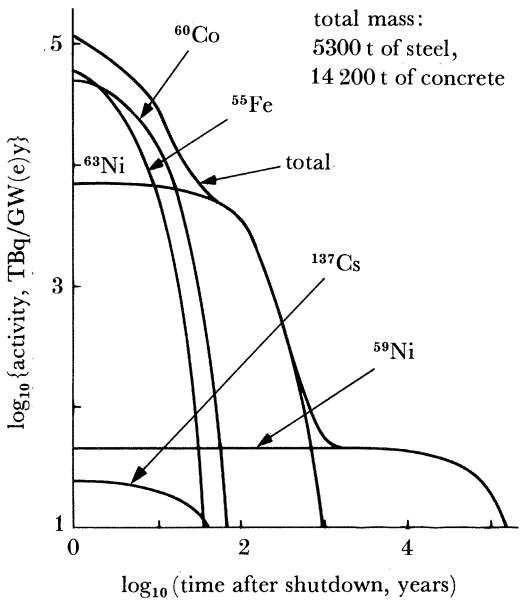

FIGURE 8. The radioactivity of PWR decommissioning wastes per GW(e) as a function of time after reactor shutdown.

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has estimated that the total mass of steels arising from decommissioning a PWR of 1 GW(e) output would amount to about 4700 t. D. Goodill (personal communication, 1985) has calculated the decay curve for these decommissioning wastes (figure 8). The advantage to be gained by delaying decommissioning until the ^{60}Co has decayed is evident. These activity figures are very low compared with the total arisings from the fuel cycle over the lifetime of a reactor, as may be seen by comparing the decay curve in figure 8 with those from the annual arisings in figures 4 and 7. Larger volumes will arise from decommissioning Magnox and AGR reactors because of the lower power density and the need to dispose of the graphite moderators, but the total activities from these components will be low.

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Discussion

T. A. Kletz (*Department of Chemical Engineering, University of Technology, Loughborough, U.K.*). I have two questions:

1. Has any thought been given to the utilization of the heat from radioactive wastes?
2. How easy is it for countries that have bought nuclear reactors to develop equipment for extracting plutonium from the spent fuel?

L. E. J. Roberts.

1. The amount of heat generated from radioactive wastes is not enough to be commercially significant except, perhaps, for small-scale, local use.
2. Methods of extracting plutonium from spent fuel are well described in the open literature. The technology would not be difficult to develop.

S. H. U. Bowie, F.R.S. (*Tanyard Farm, Clapton, Crewkerne, Somerset, U.K.*). Could Dr Roberts confirm that, if waste is reprocessed to remove uranium and plutonium, after a period of approximately 10000 years the activity of the remaining waste would be equivalent to that of a similar tonnage of natural uranium? This is most important when it comes to making waste management decisions.

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L. E. J. Roberts. If spent fuel is reprocessed to remove uranium and plutonium, the activity remaining in the high-level waste after about 10000 years is approximately the same as that associated with the original quantity of natural uranium. But the mass of the high-level waste is much less than that of the original uranium and so the comparison is not valid on a tonnage basis.

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