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Institut für Angewandte Systemtechnik und Reaktorphysik
Projekt Schneller Brüter

Radioactive Waste Management
Potentials and Hazards from a Risk Point of View

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Abstract

The problems of radioactive waste management are addressed in this paper. The magnitude of the problem is identified as primarily concerns the Federal Republic of Germany. Predictions are included of the characteristics of the wastes from the various activities of the nuclear fuel cycle, and their accumulation as a function of time until the year 2010. All important reactor types are considered.

The methods for the evaluation of risk are reviewed as well as possible extensions of the methods needed to evaluate risk from radioactive waste management strategies. For input to risk evaluation a hazard index for radioactive isotopes is established and calculated for fuel reprocessing wastes from several reactor types.

Proposed waste management strategies are reviewed and their potentials, in terms of risk reduction, are identified. An attempt is made to place the segments of the waste management strategies contributing to risk in perspective.

Further effort and a refinement of the work in the area of radioactive waste management evaluation is proposed and outlined.

31.5.1974

Die Handhabung radioaktiven Abfalls

Strategien und Risiken

Kurzfassung

In diesem Bericht werden die Probleme des radioaktiven Abfalls, wie sie sich vor allem für die BRD stellen, behandelt.

Eingeschlossen sind Voraussagen über die Zusammensetzung des Abfalls in den verschiedenen Teilbereichen des Brennstoffzyklus und die zu erwartenden Gesamtmengen bis hin zum Jahr 2010. Die Betrachtung erstreckt sich auf alle Reaktortypen.

Der Bericht bringt sowohl eine Übersicht über die Methoden der Risikoberechnung als auch eine Darstellung möglicher Erweiterungen dieser Methoden auf Abfallstrategien. Als Eingangsgröße der Risikoberechnung wird ein Gefahren-Index (hazard index) radioaktiver Isotope definiert und für den radioaktiven Abfall der Wiederaufarbeitung von Brennstoff verschiedener Reaktortypen berechnet.

Es erfolgt eine Gegenüberstellung der vorgeschlagenen Abfallhandhabungsstrategien insbesondere im Hinblick auf eine mögliche Verringerung der damit verbundenen Risiken. Außerdem wird versucht, die Teilbereiche einer Strategie zu beurteilen.

Weitere Bemühungen auf dem Gebiet der Abfallhandhabung werden vorgeschlagen und skizziert.

Table of Contents

I. Introduction

The disposal of any type of hazardous material, whether it be nuclear or non-nuclear, should involve the permanent removal of the material from man's biosphere. This can be accomplished either by converting the hazardous material to nonhazardous material or by disposing of the material in such a manner that it can not reenter man's environment. However, most present day schemes for the disposal of hazardous material can not guarantee, nor do they pretend to guarantee, permanent removal of the material. As it has almost universally been practiced to date the answer to the question of how to dispose of wastes is that "dilution is the solution of pollution". This type of solution to waste disposal is being rejected presently, with due justification, and in the near future it will be totally unacceptable due to physical and biological limitations of our planet. It is, therefore, imperative that we find and utilize waste disposal systems which can guarantee, with reasonable assurance, the safety of man's biosphere. The safety of man's biosphere refers to the prevention of any undesirable change in the physical, chemical, or biological characteristics of the air, land, and water that may or will harmfully affect human life or that of other desirable species.

The disposal of radioactive wastes has associated with it special problems not encountered with other hazardous materials. That is, the damaging influence of the radioactive material to a biological system can be accomplished without incorporation into the system, or direct contact with the system. Since the radioactivity of the material is an intrinsic property the hazard associated with it can only be neutralized through the very difficult process of nuclear transmutation. In addition, mankind is particularly sensitive to the menace of radiation. He is prepared to accept hazards from

industrial and domestic accidents and from natural disasters which he is not willing to suffer from nuclear radiation. While this may seem illogical, it is not a point to be argued against but rather should be taken into due regard when considering disposal systems for radioactive wastes.

In our considerations of radioactive waste disposal systems that follows we have logically divided the systems into three separate categories. These are

1) ultimate disposal

2) long-term disposal, and

3) storage.

A basic explanation of each of these three categories is in order. We use the words "ultimate disposal" in the sense that after disposal of the waste has been successfully accomplished it is impossible for it to return to man's biosphere. We are referring to disposal by nuclear transmutation and deep space disposal. Long-term disposal refers to disposal on the earth in such a manner that the waste is nonretrievable. While the purposed schemes claim a high degree of safety it is conceivable that under certain conditions the waste may return, in an uncontrollable manner, to man's biosphere. Two of the methods considered for long-term disposal are disposal as solids in salt mines or under the Antarctica ice cap. The distinction between long-term disposal and storage is that the wastes are retrievable from storage and therefore require perpetual care. For example, the wastes may be stored as solids in man-made vaults or as liquids in tanks.

As will be noticed, to some extent slightly more emphasis in this paper is placed on the long-term disposal concept of solidified wastes in salt deposits than the other concepts. The reason is

that this concept has been the most appealing and intensively studied to date in the BRD. Therefore, the possible appearance of such a long-term disposal facility and the safety problems associated with the concept are better defined. This does not mean that we preclude the other waste disposal concepts from serious consideration - on the contrary.

Our primary concern in this paper is the radioactive wastes from spent fuel reprocessing which represent, on a curie or hazard base, 99% of the radioactive wastes generated in the nuclear fuel cycle. The waste generated from other portions of the nuclear fuel cycle present a problem primarily because of its volume. We concern ourselves here with wastes whose dominating property is its intense radioactivity.

Finally, it must be clearly stated that this paper does not offer solutions to the problem of radioactive waste management. Rather the intent is to outline the general problem, illustrate its magnitude and importance, and serve as a preliminary report of the activities performed in this field. In addition, means by which the various waste disposal procedures can be evaluated and compared in a realistic fashion are described. The approach is from a risk point of view. This paper is also intended to serve as a guide for future efforts in this field by pointing out the important and govering aspects of the problem.

II. Definition of Problem

To assess the magnitude and importance of the problem of radioactive waste management it is necessary to start with a description of the nuclear fuel cycle, the functions of its various components, and the characteristic types of waste generation. Of equal importance is an estimate of the amount of the waste generated as a function of time.

These problems are handled here by first describing the uranium/plutonium nuclear fuel cycle and its operation with associated wastes. An estimate of the contribution of nuclear energy to the total electrical energy generation in the BRD is also provided. With these estimates the amount of waste generated is calculated.

A. Nuclear Fuel Cycle

Associated with the production of energy, whether it be from fossil or nuclear fuel, is the problem of wastes from the energy production and its burden on man's biosphere. The waste problem encompasses the entire spectrum of activities in the generation of nuclear energy as in other types of energy systems. It begins with the tailings of uranium mining and is prevalent through the steps of fuel manufacturing, reactor operation, fuel reprocessing, and finally nuclear facility decommissioning at the end of their economic life.

To assess the waste problem from such an energy generating system it is imperative that one considers the entire system as an entity in itself and not treat the individual processes separately. The reason for such a consideration is elementary in systems analysis. Since the individual processes in the energy system are intimately interconnected, a change in one process is reflected by subsequent alterations in one or more of the other processes. Our nuclear energy generating system as it is presently operating, or envisioned for the near future, is shown in Fig. 1. In the following we shall discuss each stage of the energy system and the type of waste associated with each.

(i) Uranium Mining and Milling

The principal radioisotope hazard to man within a uranium mine is radon, with its radioactive decay products. In addition to radon, mine air may contain radium and other toxic constituents of the ore such as arsenic, cobalt, vanadium, and selenium. Even after the commercial ore has been removed, the walls of the mine continue to emanate radon due to the presence of low-grade ores.

The end products of milling are uranium concentrate (U_3O_8) and "tailings". The tailings contain most of the radium originally present in the ore. Since radium is one of the most toxic of all radioactive elements, its represents a serious potential hazard. Water draining from tailings ponds usually contains more soluble radium than permissible in drinking water. It is necessary, therefore, to see that this water does not contaminate public supplies.

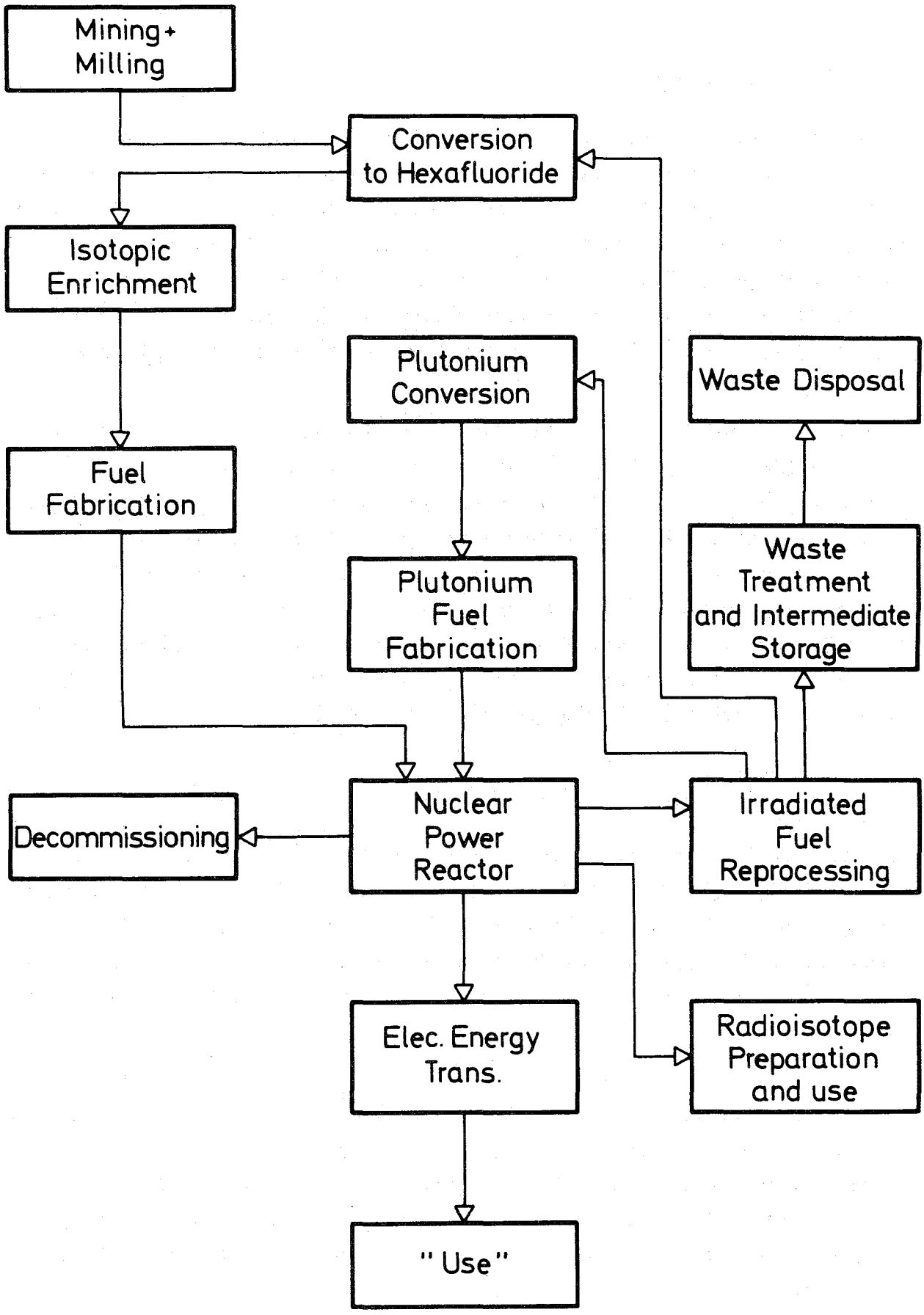


Fig.1 Nuclear Energy Generating System

It should be mentioned that wastes similar to those found in uranium mining and milling occur also in thorium extraction. The products of thorium decay are believed to be somewhat more toxic than those from uranium.

(ii) Processing of Uranium (Conversion to Hexafluoride and Isotopic Enrichment)

The wastes from these processes are very similar to those produced by the uranium mills. However, the amount of radium in the concentrate is so small that oxide processing and isotope enrichment under normal circumstances seldom produce hazardous wastes. An accident or sabotage in an isotope enrichment plant can result, however, in the formation of large quantities of fission products if a subsequent criticality accident occurs.

The waste from the production of uranium metal by reducing uranium tetrafluoride with powdered magnesium at high temperatures contains a small amount of uranium.

(iii) Fuel Fabrication

Normal wastes from fuel fabrication consist of uranium or uranium oxide in the form of dust, metal turnings and scrap, and cladding materials or other objects contaminated with uranium. Clearing materials, protective clothing, filters, and wash water must also be considered.

Since uranium is valuable, any large amount will be recovered, but recovery procedures themselves may produce wastes. Drainage from fuel fabrication plants contain suspended and dissolved uranium.

(iv) Nuclear Power Reactor Operation

Wastes arising in normal reactor operation vary considerably with different designs. It is within the reactor itself that 99% of the

total radioactivity of all wastes, associated with the generation of nuclear energy, are produced (fission products). Under normal conditions this waste is under control and is held within the fuel element by the fuel cladding. However, some fission product gases may be vented or significant quantities released by rupture of the fuel cladding.

Normal operation maintenance will produce radioactive items such as pumps, valves, piping, tanks and parts of the internal reactor mechanism. The decontamination center will produce used cleaning fluids, mops and rags. In all reactors unwanted impurities occur in the coolant, resulting generally in induced reactivity. To combat the buildup of radioactivity in the coolant, part of the coolant is bled to waste and replaced with pure coolant. In the use of ion-exchange columns to cleanse the coolant highly radioactive resins, and strong salt solutions arising from the regeneration of the resins, are produced.

Radioactive gases are emitted from reactor stacks even if the effluent is passed through the best of filters and scrubbing systems. The concentration of the isotopes is dependent on the reactor type but the effluents generally contain ^{85}Kr, ^{131}I, ^{133}Xe and ^{3}H. An accident, or act of sabotage, can result in the release of large quantities of radioactive nuclides.

(v) Irradiated Fuel Reprocessing

The type of waste produced depends upon the design of the fuel, the method used for dissolving the fuel, and the particular processing system employed. Volatile fission products are released at the time of dissolution of the fuel. Tritium is largely oxidized and stays with the process water, going out with the low-level aqueous effluent. The volatile radioisotope iodine is largely scrubbed out of the process off-gas although some is released to the environment. The radioactive noble gases are released with the gaseous plant effluent. The low-level liquid waste contains, in addition to tritium, small concentrations of fission products which are carried over by entrainment in the various evaporation steps.

In present day light water reactors approximately 0,9 kg of non-gaseous fission products result from each 1000 MWD of thermal energy produced. In addition, transuranium and transplutonium elements (americium, curium, neptunium, etc.) and plutonium (as a waste loss) are present in the waste to the extent of approximately 0,02 kg/1000 MWD of thermal energy.

(vi) Waste Treatment and Intermediate Storage

Interim storage of radioactive waste is usually done with the waste in liquid form in steel tanks. Releases to the environment here result from leakage during handling or rupture of the tanks. The solidification of the wastes by pot calcination, spray solidification, phosphate-glass solidification, or fluidized-bed solidification do not return 100% of the fission product activity to the principal solid product from the process. A low-level aqueous waste liquid remains, which must be treated in a separate system, and some of the radioactive elements are released with an effluent gas stream.

The potential for sabotage at this point in the system is great, resulting in the release of millions of curies of fission products.

(vii) Waste Disposal

This point in the energy system will be discussed at great lengths in this paper and does not need to be expanded upon here. It should be mentioned that presently there is no agreed upon method for disposal of high-level radioactive wastes.

(viii) Plutonium Conversion and Plutonium Fuel Fabrication

The waste problems that are encountered here are similar to these presented under points (ii) and (iii) except that plutonium replaces uranium in the waste stream with its associated greater hazard. As a result the volume of waste which must be properly handled increases tremendously over that resulting in the fabrication of uranium fuel elements.

(ix) Radioisotope Preparation and Use

Hospitals and biological laboratories give rise to special types of waste mainly containing radionuclides with short half-lives. The most hazardous

wastes result from rupture of large sealed sources. The highest level wastes arising from the isotope production industry consist of spent multi-curie sources.

(x) Decommissioning

The waste problems in decommissioning of a reactor and other facilities of the nuclear fuel cycle at the end of their economic lives are the induced radioactive structural materials and possibly reactor coolant, as well as contaminated parts. This particular area in the energy system has received very little attention to date, but will require concentrated efforts to solve the possible problems present here.

(xi) Transportation

Transportation of radioactive material is involved at every stage of the nuclear energy generating system. For example, for a 1000 MWe pressurized water reactor the transportation requirements in terms of tons per year are /1/:

85,500	tons	ORE
162	"	U_3O_8
203	"	UF_6
53	"	ENRICHED UF_6
36	"	FUEL
36	"	SPENT FUEL
4	"	SOLID WASTE

Up to the point where spent fuel is shipped from the reactor, the transportation problems are relatively simple. However, an accident with spent fuel in transportation can represent significant hazards. If instead our entire system is considered as a "nuclear park" the transportation problem and its associated sabotage potential, are relaxed, but other problems may arise.

The U.S. Atomic Energy Commission has recently compiled a comprehensive report /2/ on the nuclear fuel cycle and its operation and wastes. Included is all pertinent data such as, energy, water, air, and materials (or raw products) needed in the fuel cycle, normalized to the needs of a "model" 1000 MWe LWR power plant.

The areas of the fuel cycle involving potential environmental impacts of significance are:

1) routine effluents of radioactivity from nuclear power plants and fuel reprocessing facilities

2) transportation of irradiated fuel from reactor to reprocessing facility, and radioactive wastes from reprocessing facility to disposal area

3) the disposal of radioactive wastes.

B. Nuclear Energy Predictions (1970 - 2010 in the BRD)

For the purpose of predicting energy generation over a period of 40 years one needs to understand the parameters which influence the rate of energy generation. At best most of these parameters are presently poorly understood and some totally unknown. Most so-called energy demand studies are basically just extrapolations of historical trends in energy consumptions. There is, however, no established reason to believe that future energy needs are necessarily determined by extrapolations of historical energy consumptions trends. In a review by the U.S. House Committee on Interior and Insular Affairs /3/ of 35 federal agency reports on the estimate of future energy consumption it is stated that, in general, in these 35 predictions all used similar projection techniques and data, similar assumptions were made, and not surprisingly, similar results were obtained. One notable conclusion was, however, that most recent studies tend to have higher forecast values, reflecting the higher energy consumption rates of the last ten years which earlier studies were unable to incorporate into their calculations of trends. This has also been found to be true in projections of nuclear power /4/.

As a point of reference electrical energy today accounts for approximately 25% of the total energy consumption in the BRD /5/. The remaining 75% of the total energy consumption is divided just about evenly between residential and commercial, industrial, and transportation. The average growth rate per year of total energy consumption from 1950 to 1970 was 4.9% in the BRD. The average growth rate per year of electrical energy production from 1960 to 1970 was 7.9%. This leads to the obvious conclusion that the portion electrical energy in the total energy consumption will be increasing from its present 25%. Predictions for the BRD show that by the end of the century electrical energy will have risen to about 50% of the total energy production /5/, /6/.

If one estimates that 80 to 85% of the electrical energy generation will be from nuclear power by the year 2000, slightly more than 40% of the total energy consumption will be from nuclear sources. There is a very good possibility that by the year 2000 the high-temperature reactors will be

providing process heat for the chemical and iron industries to the extent of approximately 3% of the total energy consumption. This would bring the nuclear portion to about 43% of the total energy production.

The above remarks serve to put the energy production and its prediction into a proper frame of reference. It is not the goal of this work to predict what the future energy consumption will be, but rather to assess ultimately the possible environmental impact of the nuclear fuel cycle wastes. For this reason the results of a study /7/ performed for this same purpose are used in this work. The results were calculated as a band into which the future energy consumption will possibly fall. The installed electrical generating capacity, total and nuclear, from /7/ are shown in Fig. 2. For details concerning the basis of the curves reference should be made to Ref. /7/.

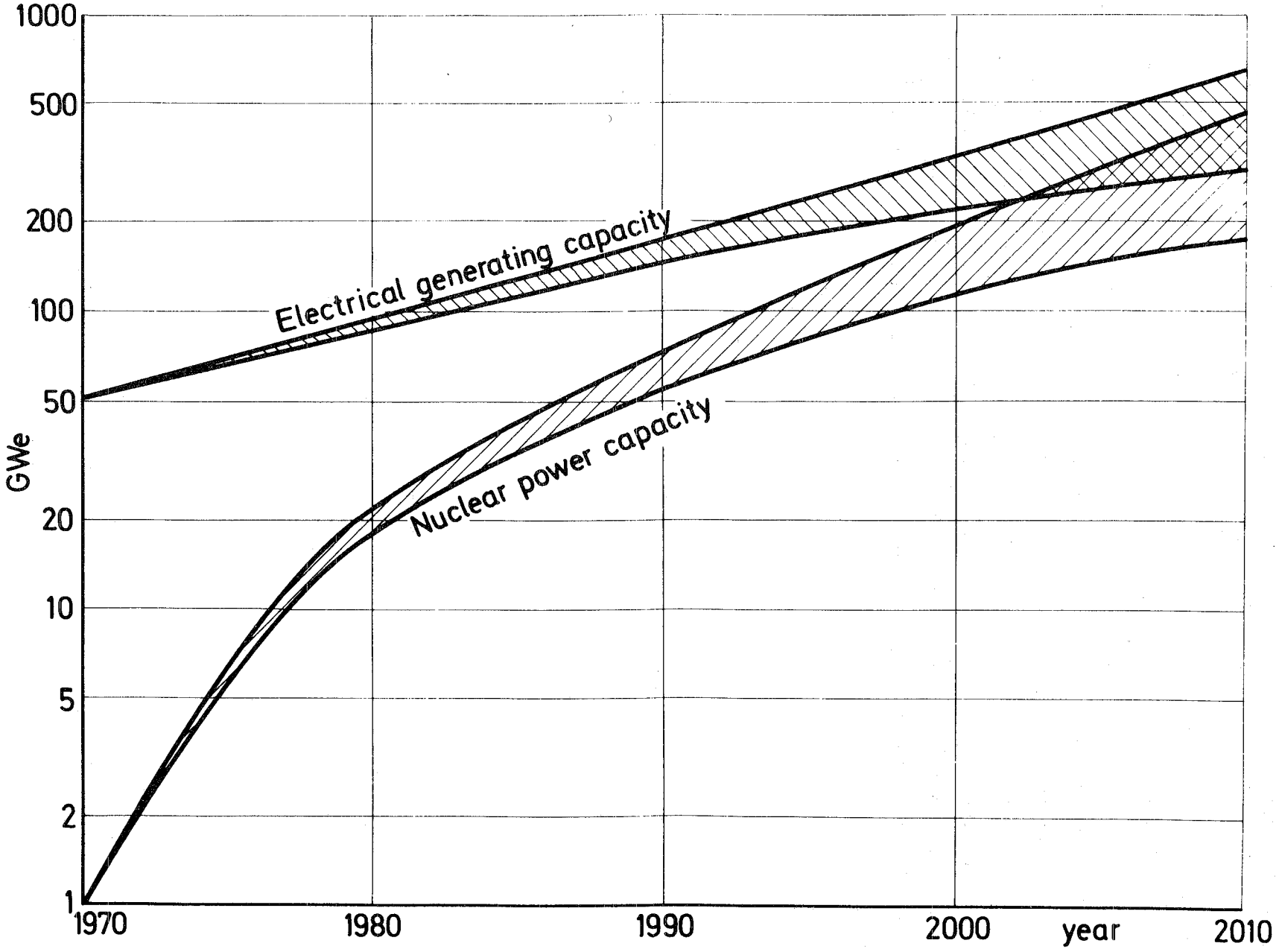


Fig. 2 Installed Electrical Generating Capacity and Nuclear Power in the BRD

C. Radioactive Waste Production

In the various components, or stages , of the nuclear fuel cycle as described in section A, there are three main components which generate significant amounts of radioactive wastes. The three main components are fuel fabrication (both uranium and plutonium), nuclear power reactor operation, and irradiated fuel reprocessing. These three components are discussed separately below, in particular as they apply to the situation in the BRD. For the calculations of the radioactive waste production the nuclear energy estimates, as presented in Fig. 2 of this report from Ref. /7/, were used. Rather than using the band of the estimates given, the mean or middle value was taken, as given in Table 1 below. In addition, the share of the light water and fast breeder reactors in the total installed nuclear capacity is given. For the calculation of this division of the reactors it was assumed that the light water reactor would cover the installed capacity totally up to 1987, after which time fast breeder reactors are built at a rate determined by the availability of plutonium in the BRD from its own nuclear industry.

The amount of fuel fabrication and reprocessing work required for this reactor strategy was calculated in Ref. /7/. The results are given here in Fig. 3 and 4. In addition, the yearly production of plutonium from both the light water and fast breeder reactors is shown in Fig. 5.

Table 1: Estimated Growth of Nuclear Power in the BRD (GWe)

Year	Nuclear Installed Capacity	LWR[1)]	LMFBR[2)]
1970	1	1	
1975	9	9	
1980	20	20	
1985	36	36	
1990	63	60	3
1995	105	91	14
2000	155	123	32
2005	232	173	59
2010	316	216	100

1) LWR: Light water thermal reactors consisting of 50% pressurized water reactors and 50% boiling water reactors.

2) LMFBR: Liquid metal fast breeder reactor, sodium cooled and oxide plutonium fuel.

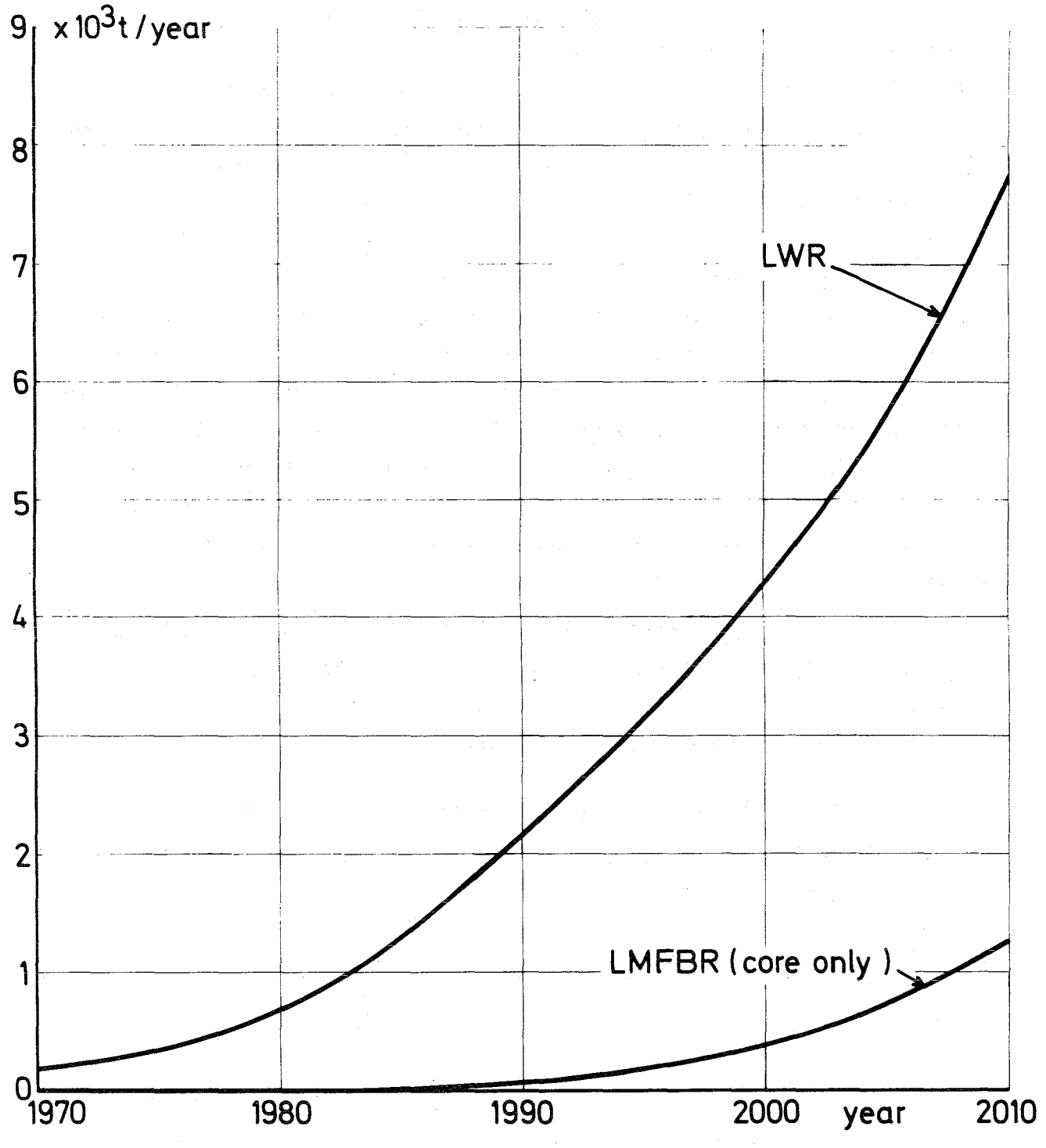


Fig. 3 Annual Fuel Fabrication Requirements

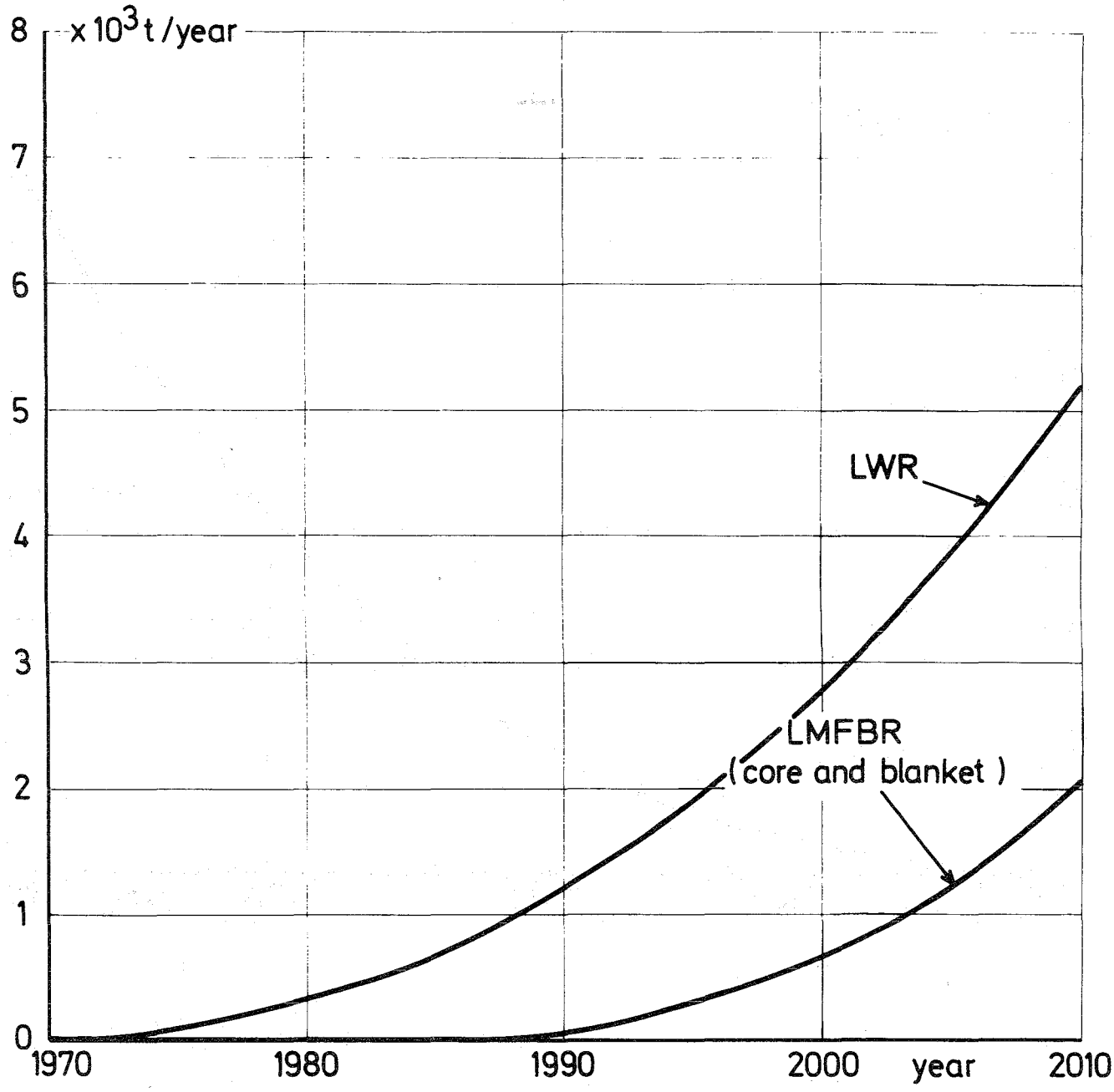


Fig. 4 Annual Fuel Reprocessing Requirements

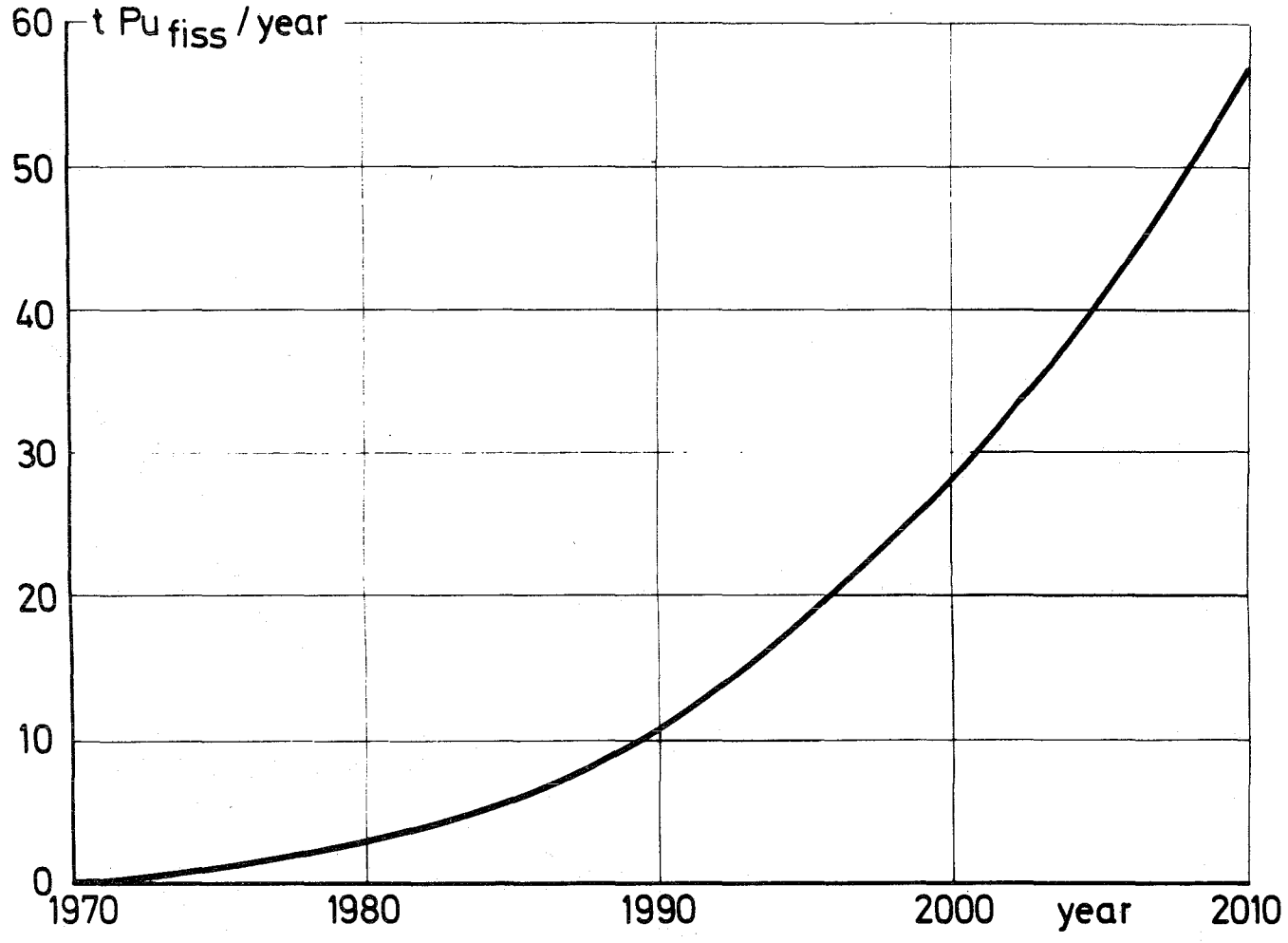


Fig. 5 Annual Plutonium Production

The reactor data used in the calculations are representative of current designs and are listed in Ref. /7/. In these calculations it was assumed that 50% of the light water reactors are pressurized water reactors and 50% boiling water reactors.

C1. Fuel Fabrication Wastes

The solid wastes generated at fuel fabrication plants consist primarily of paper, plastic, gloves, and other wastes contaminated with low-level uranium or plutonium. The liquid wastes generated are solidified and package for disposal. The estimated low-level wastes from the fabrication of fuel are

LWR:	5.9 m^3/ton fuel	(uranium fuel)
	11.8 m^3/ton fuel	(plutonium recycle fuel)
LMFBR:	9.7 m^3/ton fuel	(plutonium fuel)

C2. Nuclear Power Plant Wastes

During the operation of nuclear power reactors gaseous, liquid, and solid wastes are produced. However, the amount and composition varies considerably with reactor type and even among reactors of the same type. A significant portion of the wastes generated at the nuclear power plant are released to the environment immediately, or after a short delay time. The radioactivity in the released wastes is so diluted that it is not believed to represent a hazard to the general population.

The gaseous wastes from the reactor result from the leakage of gaseous fission products from the fuel. In some incidents radioactive gas removal systems are used to collect the escaping gas. The amount (volume) collected over a year of operation is generally not large. For example, for the LMFBR it is estimated that in one year less than 50 liters of ^{85}Kr will be collected /8/.

The low- and medium-level liquid wastes generated at the reactor are treated to remove radioactivity by methods such as evaporation or ion exchange. The concentrate containing the radioisotopes are solidified and stored as solid wastes.

Solid wastes typically consist of filters, laboratory and liquid waste treatment residues, contaminated tools, parts and wastes such as plastic bags, foot covers, paper towels and protective clothing. This waste is compacted, or incinerated, and packaged for burial.

The amount of solid or semi-solid waste, from light water reactors after dewatering but before treatment and packaging, have a rather wide range of values. Typically they are as given in the following table.

Table 2 Light Water Reactor Solid Wastes from Reactor Operation /9/

Waste Type	Vol. (m^3/yr)	Activity (curie/m^3)
BWR:		
spent resins	5.6 - 11.2	< 7
sludges (condensate clean-up filters, clean-up-systems)	22 - 64	3.5 - 70
PWR:		
spent resins	4.8 - 7	175 - 3500
evaporator bottoms	1.4 - 4.2	< 35

The medium-level wastes are generally fixed in concrete and packaged in drums, each of about 55 gallons (US), or 0.21 m^3, capacity. The low-level wastes are not fixed in any inert material. Therefore, for 1000 MWe reactor plants the following volumes of low- and medium-level package wastes are generated

BWR: 140 - 280 m^3/year

PWR: 50 - 80 m^3/year

Since to date there has not existed a large commercial fast breeder reactor, it is difficult to estimate reliably the volume of wastes. The estimates for a 1000 MWe LMFBR /8/ are, for packaged solid low- and medium-level wastes,

LMFBR: 28 - 60 m^3/year.

For the high-temperature gas cooled reactors of the 1000 MWe size the volume of wastes would be expected to be, extrapolated from the Fort St. Vrain HTGR reactor

HTGR: $\sim$ 33 m^3/year.

C3. Fuel Reprocessing Wastes

The bulk of the very hazardous radioactive wastes generated in the nuclear fuel cycle are produced at the reprocessing plant. The wastes are generated in gaseous, liquid, and solid form. Much of the gaseous waste is presently released from the facility almost immediately after the spent fuel is reprocessed. The gaseous wastes consist primarily of the noble fission product gases. Solid wastes from the reprocessing plant include fuel cladding hulls and other fuel element parts, and low-level wastes similar to that generated at the reactor (laboratory wastes, small tools, gloves and clothing).

The reprocessing of the fuel is a liquid process from which 99% of the uranium and plutonium present is removed. After some volume reduction the high-level wastes, which contain 99% of the fission products and actinides of the original spent fuel element, are stored in stainless steel tanks before further treatment. The medium-level wastes generated are the spent contaminated chemical agents used in the reprocessing, off-gas scrubbing agents and solutions from decontamination operations. In addition, it is estimated that something of the order of 50% of the plutonium loss to waste during the fuel reprocessing ends up in the medium-level wastes.

The volumes of wastes from fuel reprocessing, per ton of spent fuel, are given in Table 3. From the estimated growth of nuclear energy in the BRD, given in Table 1, the estimated production of high-level wastes are given in Table 4.

Table 3: Wastes from Fuel Reprocessing Plant per Ton Fuel

	LWR	LMFBR
High-level liquid, m^3	0.6-1.0	0.6-1.0
Convertion of high-level liquid waste to solid, liters	80-100	80-100
Cladding hulls and other fuel element parts, m^3	0.3-0.6	0.5-1.0
Medium-level liquid (concentrates), m^3	$\sim$ 3.0	$\sim$ 3.0
Low-level solid wastes, m^3	0.6-4.0	2.4-15.0

Table 4: Estimated Production of High-Level Fuel Reprocessing Wastes in the BRD

	1980	1990	2000	2010
Installed Capacity, GWe	20	63	155	316
Amount of Fuel Reprocessing[1], tons/years:				
LWR	356	1232	2838	5224
LMFBR		42	609	2075
Accumulated Amount of Fuel Reprocessing, tons:				
LWR	1394	8915	28924	68585
LMFBR		105	2886	15556
Volume of Waste Generated as Liquid [2]				
Annually, m^3/year	356	1274	3447	7299
Accumulated, m^3	1.39×10^3	9.92×10^3	3.18×10^4	7.41×10^4
Volume of Waste Generated as Solid[3]				
Annually, liters/year	8.40×10^3	5.69×10^4	1.76×10^5	3.72×10^5
Accumulated, liters	1.93×10^4	3.25×10^5	1.58×10^6	4.26×10^6

[1] Reprocessing of fuel at 150 days after discharge from reactor

[2] 1.0 m^3/ton fuel at 34,000 MWD/T burnup

[3] 80 liters glass/ton fuel at 34,000 MWD/T five years after reprocessing

The characteristics of the high-level wastes were calculated for a number of the most interesting and important reactors, namely:

1) PWR with an equilibrium uranium fuel cycle
2) PWR with plutonium recycle fuel
3) LMFBR with plutonium from an LWR reactor industry
4) LMFBR with equilibrium fast reactor plutonium
5) HTGR (high temperature gas cooled reactor) with an equilibrium ^{235}U fuel cycle
6) THTR (high temperature gas cooled reactor) with ^{233}U recycle fuel and ^{235}U makeup.

In the LMFBR results it is assumed that the core and blanket are mixed proportionally in the fuel reprocessing facility. In addition, it was assumed that the PWR spent fuel wastes could be taken as representative of all LWR wastes. Since BWR's do not presently achieve a burnup reached in PWR's, the assumption leads to an overestimate of the total LWR wastes by approximately 10%. If BWR's obtain in the future a burnup equivalent with PWR's then our assumption will be nullified.

For the calculations a set of fuel cycle programs /11/ were used to obtain global descriptions and average parameters of the reactors. The total isotopic composition and decay heats of the spent fuels were calculated then by the ORIGEN program /12/. The calculated characteristics of the high-level wastes per ton of spent fuel are given in Tables 5 to 10. The time periods in the tables are in reference to the time the fuel was reprocessed, which in all cases was assumed to occur 150 days after the fuel was discharged from the reactor. It was assumed that the plutonium loss to the waste in all cases was 1.0% of the plutonium originally present in the spent fuel elements. Present reprocessing losses for plutonium are somewhat larger, ranging anywhere from 1.0% to 6.0%.

The data used in the reactor calculations are representative of present and near-future generation reactors. The light water reactor data were obtained from KWU /13/ and are representative of the type of reactors KWU is presently offering on the light water reactor market. For the LMFBR data the 2000 MWe reference design of Interatom and Projekt Schneller Brüter /14/ was used. Finally, for the high temperature gas cooled reactors use was made of the Oak Ridge prepared "Evaluation of High-Temperature Gas-Cooled Reactors" /15/.

Table 5: Characteristics of PWR(1)[1] Spent Fuel Wastes (per metric ton fuel)

Time since reprocessing[2] (years)	1.0	3.0	5.0	10.0	30.0
Total activity, MCi	1.69	0.704	0.459	0.320	0.186
Total heat-generation rate, Kw	8.21	3.16	1.77	1.06	0.58
Radioisotopes, Ci					
^{90}Sr	7.64+4	7.27+4	6.92+4	6.12+4	3.74+4
^{137}Cs	1.07+5	1.02+5	9.77+4	8.71+4	5.48+4
^{129}I	3.86-2	3.86-2	3.86-2	3.86-2	3.86-2
^{85}Kr	1.05+4	9.24+3	8.13+3	5.90+3	1.63+3
^{3}H	6.72+2	6.02+2	5.38+2	4.06+2	1.32+2
^{238}Pu[3]	1.06+2	1.24+2	1.23+2	1.18+2	1.02+2
^{239}Pu	3.24	3.24	3.24	3.24	3.24
^{240}Pu	5.13	5.66	6.15	7.22	9.93
^{241}Pu	1.01+3	9.22+2	8.39+2	6.61+2	2.56+2
^{241}Am	1.63+2	1.66+2	1.68+2	1.73+2	1.81+2
^{243}Am	2.02+1	2.02+1	2.02+1	2.02+1	2.02+1
^{242}Cm	4.07+3	1.90+2	1.60+1	7.67	6.99
^{244}Cm	2.68+3	2.49+3	2.30+3	1.90+3	8.84+2

[1] PWR(1)-pressurized light water reactor with equilibrium uranium fuel cycle at 34 000 MWD/T burnup (initial enrichment 3.3%)

[2] Reprocessing of fuel 150 days after discharge from reactor

[3] 1.0% plutonium loss to waste stream

Table 6: Characteristics of PWR(2)[1] Spent Fuel Wastes (per metric ton fuel)

Time since reprocessing[2] (year)	1.0	3.0	5.0	10.0	30.0
Total activity, MCi	1.75	0.722	0.468	0.325	0.187
Total heat-generation rate, Kw	9.24	3.77	2.38	1.55	0.83
Radioisotopes, Ci					
^{90}Sr	6.83+4	6.50+4	6.19+4	5.47+4	3.34+4
^{137}Cs	1.08+5	1.03+5	9.86+4	8.78+4	5.53+4
^{129}I	4.14-2	4.14-2	4.14-2	4.14-2	4.14-2
^{85}Kr	9.50+3	8.35+3	7.35+3	5.33+3	1.48+3
^{3}H	7.29+2	6.51+2	5.82+2	4.39+2	1.42+2
^{238}Pu[3]	2.31+2	2.76+2	2.75+2	2.65+2	2.29+2
^{239}Pu	3.29	3.29	3.29	3.29	3.29
^{240}Pu	7.97	1.22+1	1.61+1	2.46+1	4.61+1
^{241}Pu	1.40+3	1.27+3	1.16+3	9.15+2	3.57+2
^{241}Am	2.76+2	2.79+2	2.82+2	2.88+2	2.98+2
^{243}Am	8.88+1	8.88+1	8.88+1	8.88+1	8.88+1
^{242}Cm	1.03+4	4.77+2	3.74+1	1.64+1	1.49+1
^{244}Cm	2.13+4	1.97+4	1.83+4	1.51+4	7.01+3

[1] PWR(2)-pressurized light water reactor with plutonium recycle fuel at 34 000 MWD/T burnup, 19% of fissile charge is plutonium

[2] Reprocessing of fuel 150 days after discharge from reactor

[3] 1.0% plutonium loss to waste stream

Table 7: Characteristics of LMFBR(1)[1] Spent Fuel Wastes (per metric fuel ton)

Time since reprocessing[2] (years)	1.0	3.0	5.0	10.0	30.0
Total activity, MCi	2.29	0.832	0.471	0.287	0.161
Total heat-generation rate, Kw	10.35	3.11	1.50	0.84	0.51
Radioisotopes, Ci					
^{90}Sr	4.48+4	4.27+4	4.06+4	3.59+4	2.19+4
^{137}Cs	1.13+5	1.08+5	1.03+5	9.15+4	5.77+4
^{129}I	3.59-2	3.59-2	3.59-2	3.59-2	3.59-2
^{85}Kr	7.48+3	6.58+3	5.79+3	4.20+3	1.16+3
^{3}H	8.93+2	7.98+2	7.13+2	5.38+2	1.74+2
^{238}Pu[3]	3.27+2	3.56+2	3.53+2	3.42+2	3.02+2
^{239}Pu	4.32+1	4.32+1	4.32+1	4.32+1	4.32+1
^{240}Pu	4.92+1	4.94+1	4.95+1	4.99+1	5.07+1
^{241}Pu	5.74+3	5.22+3	4.75+3	3.75+3	1.45+3
^{241}Am	2.50+3	2.51+3	2.52+3	2.53+3	2.54+3
^{243}Am	4.26+1	4.26+1	4.26+1	4.26+1	4.25+1
^{242}Cm	6.88+3	3.77+2	8.57+1	7.05+1	6.43+1
^{244}Cm	8.84+2	8.19+2	7.59+2	6.26+2	2.91+2

[1] LMFBR(1): sodium cooled fast breeder reactor fueled with plutonium from an LWR industry, core and blanket mixed proportionally for reprocessing, burnup ~ 34 000 MWD/T.

[2] Reprocessing of fuel 150 days after discharge from reactor

[3] 1.0% plutonium loss to waste stream

Table 8: Characteristics of LMFBR(2)[1] Spent Fuel Wastes (per metric ton fuel)

Time since reprocessing[2] (year)	1.0	3.0	5.0	10.0	30.0
Total activity, MCi	2.32	0.845	0.479	0.292	0.164
Total heat-generation rate, Kw	10.48	3.16	1.56	0.95	0.59
Radioisotopes, Ci					
^{90}Sr	4.56+4	4.34+4	4.13+4	3.65+4	2.23+4
^{137}Cs	1.14+5	1.09+5	1.04+5	9.30+4	5.86+4
^{129}I	3.64-2	3.64-2	3.64-2	3.64-2	3.64-2
^{85}Kr	7.60+3	6.69+3	5.88+3	4.27+3	1.18+3
^{3}H	9.06+2	8.10+2	7.23+2	5.46+2	1.77+2
^{238}Pu[3]	4.01+1	4.86+1	4.86+1	4.75+1	4.32+1
^{239}Pu	4.65+1	4.65+1	4.65+1	4.65+1	4.65+1
^{240}Pu	4.88+1	4.88+1	4.88+1	4.88+1	4.89+1
^{241}Pu	2.15+3	1.95+3	1.77+3	1.40+3	5.42+2
^{241}Am	7.55+2	7.59+2	7.63+2	7.69+2	7.74+2
^{243}Am	7.o8	7.o8	7.o8	7.o8	7.o8
^{242}Cm	1.90+3	1.03+2	2.30+1	1.88+1	1.71+1
^{244}Cm	1.47+2	1.36+2	1.26+2	1.04+2	4.83+1

[1] LMFBR(2): sodium cooled fast breeder reactor fueled with fast breeder equilibrium plutonium; core and blanket mixed proportionally for reprocessing; burnup ~ 34 000 MWD/T

[2] Reprocessing of fuel 150 days after discharge from reactor

[3] 1.0% plutonium loss to waste stream

Table 9: Characteristics of HTGR[1] Spent Fuel Wastes (per metric ton fuel)

Time since reprocessing[2] (year)	1.0	3.0	5.0	10.0	30.0
Total activity, MCi	2.42	1.22	0.908	0.684	0.399
Total heat-generation rate, Kw	11.2	5.20	3.47	2.26	1.26
Radioisotopes, Ci					
^{90}Sr	1.93+5	1.84+5	1.75+5	1.55+5	9.93+4
^{137}Cs	2.02+5	1.93+5	1.84+5	1.64+5	1.03+5
^{129}I	8.72-2	8.72-2	8.72-2	8.72-2	8.72-2
^{85}Kr	3.92+4	3.45+4	3.03+4	2.20+4	6.09+3
^{3}H	8.29+2	7.41+2	6.62+2	4.99+2	1.62+2
^{232}U[3]	3.20	3.14	3.08	2.94	2.42
^{238}Pu	1.06+2	1.07+2	1.05+2	1.01+2	8.67+1
^{239}Pu	0.137	0.137	0.137	0.138	0.142
^{240}Pu	0.322	0.590	0.839	1.38	2.76
^{241}Pu	9.11+1	8.29+1	7.54+1	5.95+1	2.32+1
^{241}Am	2.34+1	2.36+1	2.38+1	2.42+1	2.46+1
^{243}Am	5.62	5.62	5.62	5.62	5.62
^{242}Cm	4.59+2	2.13+1	1.64	0.702	0.640
^{244}Cm	1.36+3	1.26+3	1.17+3	9.65+2	4.49+2

[1] HTGR - high temperature reactor with no ^{233}U recycle at 65 000 MWD/T

[2] Reprocessing of fuel at 150 days after discharge from reactor

[3] 1.0% plutonium and uranium loss to waste stream

Table 10: Characteristics of THTR[1] Spent Fuel Wastes (per metric ton fuel)

Time since reprocessing[2] (years)	1.0	3.0	5.0	10.0	30.0
Total activity, MCi	2.41	1.21	0.910	0.691	0.404
Total heat-generation rate, Kw	10.7	4.91	3.33	2.25	1.28
Radioisotopes, Ci					
^{90}Sr	1.97+5	1.87+5	1.78+5	1.58+5	9.62+4
^{137}Cs	2.04+5	1.95+5	1.86+5	1.66+5	1.03+5
^{129}I	8.97-2	8.97-2	8.97-2	8.97-2	8.97-2
^{85}Kr	3.98+4	3.50+4	3.08+4	2.23+4	6.18+3
^{3}H	8.31+2	7.42+2	6.63+2	5.00+2	1.62+2
^{232}U[3]	1.78	1.75	1.71	1.63	1.35
^{238}Pu	8.98+1	8.92+1	8.79+1	8.45+1	7.24+1
^{239}Pu	9.30-2	9.31-2	9.32-2	9.34-2	9.42-2
^{240}Pu	1.34-1	1.73-1	2.10-1	2.89-1	4.90-1
^{241}Pu	5.17+1	4.70+1	4.28+1	3.37+1	1.31+1
^{241}Am	1.27+1	1.28+1	1.30+1	1.32+1	1.34+1
^{243}Am	1.39	1.39	1.39	1.39	1.38
^{242}Cm	1.68+2	7.86	6.98-1	3.54-1	3.23-1
^{244}Cm	1.99+2	1.84+2	1.71+2	1.41+2	6.55+1

[1] THTR - high temperature reactor with ^{233}U recycle fuel at 65 000 MWD/T burnup

[2] Reprocessing of fuel 150 days after discharge from reactor

[3] 1.0% plutonium and uranium loss to waste stream

III. Evaluation of Societal Risk

In recent years there has been a wide spread growth of public awareness and concern about the impact of large-scale technological activities on man's environment. The awareness and concern on the part of the public has been in part responsible for the attempt to quantify risks and benefits involved in the operation of a particular technology. Risk can be said to be an expression of the probability of a specific outcome occurring due to a specific cause, e.g. the probability of suffering a fatal accident while driving a car.

The analysis of risk can be decomposed into two broad and separate areas. To discuss the ideas involved it is instructive to follow the illustrative flow chart for risk analysis shown in Fig. 6.

The first area of the analysis of risk is what is called the "elements of risk analysis". This area lends itself to mathematical treatment. In this area the methods for calculating the risk of an activity are studied. For example, an attempt is made to calculate the probability of an undesired event occurring, the magnitude of the subsequent release of hazardous materials, the degree of interaction of the released material with the population, and finally the effect of the material on the well-being of the population. Briefly, the analytical approach to the analysis involves the following steps:

- Identification of "Events" of interest (i.e., events associated with the release of radioactive material).

- Development of Boolean expressions which describe the circumstances (subsystem or element failures) under which these events can occur - perhaps supplanted by stylized pictorial representations (e.g. fault trees).

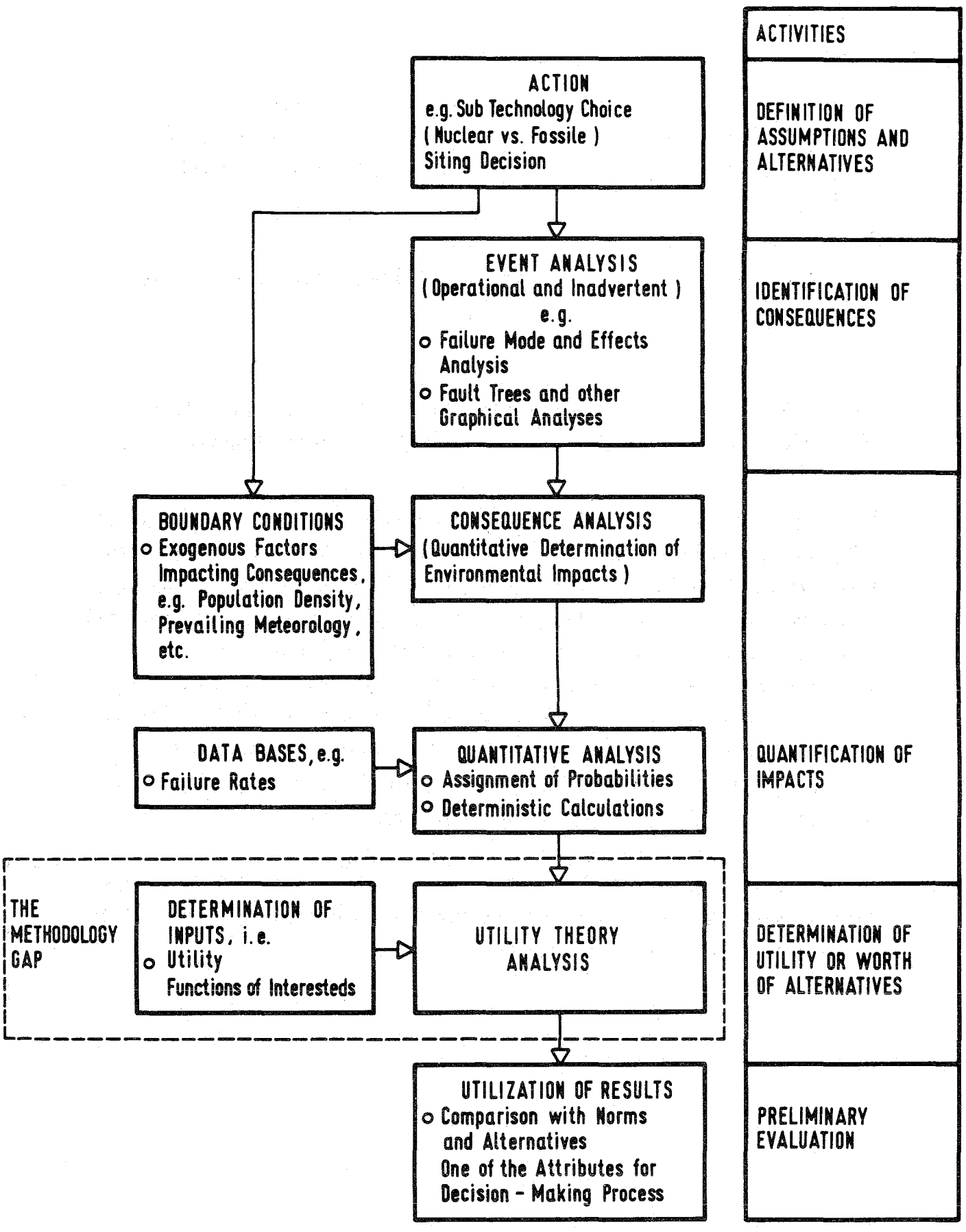


FIG. 6 ILLUSTRATIVE FLOW CHART : RISK ANALYSIS

- Development of models which evaluate the consequences of these events (e.g. mortality risks) in terms of the magnitude of release and exogenous factors (e.g. weather patterns, population densities, etc.).

- Evaluation of the likelihood of these consequences by exercising the models with data.

This type of analysis has been extensively employed by Otway /17/ and Otway et al. /19/ to analyze the safety of reactor systems.

The outputs of the analysis, as outlined above, are then used to determine the acceptability of the system. The activities entailed are the last two shown in Fig. 6, the illustrative flow chart. This is an area which may be termed "components of risks", the second area in risk analysis. Here one attempts to find the parameters important in risk acceptance, and the relationship between risk acceptance and benefits derived. For example, a few such important parameters between risk acceptance and benefits are: magnitude and frequency of accidents, voluntary verses involuntary. In other words, one attempts to place the calculated risk from the system in some perspective.

The two areas of risk analysis are discussed below in somewhat greater detail. Emphasis is given to the approach of risk evaluation of radioactive waste management procedures.

A. Components of Risks

A number of attempts have been made, most notably by Starr /16/, to find relationships between public risk acceptance and benefits derived. Starr found that the acceptance (measured by participation) of various technologies increased as the associated risk decreased. He postulated that the value of the statistical risk of death from disease to be the "psychological yard stick" by which people subjectively establish the acceptability of other risks. Otway /17/, in a study of the probability of accidental death from various causes, concluded that people intuitively seem to be unaware of risks at a mortality risk level of 10^{-6} per person per year. He therefore postulated as an acceptable maximum risk of 10^{-7} per person per year for a person living nearest to nuclear power reactors.

In a nutshell, what one does is to search for the parameters important in public risk acceptance and their associated levels of risk and then attempt to design the facilities of the technology so that their imposed risk falls below the specified limits. The assumption involved is that the society will accept the "new" technology. In other words, they will apply their preestablished "yardstick" in judging the risks of the technology. However, if the technology is new and the safety has not been sufficiently well demonstrated, and/or the credibility of the risk estimates are questioned, the society may readjust its yardstick considerably when judging the acceptance of the technology. This has certainly been the case with nuclear power.

A convenient yardstick to use in making comparisons, or seeking points of references, of nuclear power systems and associated facilities are the naturally existing background radiation fields or radiation doses one receives from medical diagnosis and treatment. For radioactive waste disposal schemes one could compare the potential hazards of the wastes to that of naturally existing radioactive ores, as uranium and thorium ore deposits. Since most waste disposal schemes call for burial of the waste the analogy is close. The establishment that the wastes, due to the disposal scheme, present a hazard of the same order

of magnitude, or smaller, than common uranium ore is no assurance, however, that society will accept the risk. The two possibilities existing are to gain public acceptance by exhaustively demonstrating safety or by modifying the disposal system so as to be acceptable.

An important question in this context is that of social rate of discount. The production of energy by fission produces wastes that will remain potentially dangerous for many hundreds of years. We will, therefore, pass our wastes from energy produced today to generations of the future. The value we intuitively place on the life of someone in a future generation reflects our social rate of discount. For example, the value of life to our children's children, yet to be born, is quite high, but in the more distance future the value declines rapidly. We will have to address the question of social rate of discount when considering the potential hazards of waste.

B. Elements of Risk Analysis

As explained above the elements of risk analysis concerns itself with the specific calculation, through various means, of the risk to a person in the population of meeting a given fate. The approach is to calculate from mathematical principles, or statistical data when it is available, the probability of a specific event occurring. This approach can be classified broadly under the heading of "design under risk". In this sense risk is understood to mean chance, as in gambling. In other words, it is a condition under which it is possible to calculate in one manner or another, the probability of all possible events occurring. It is, however, also possible that there may be instances when one does not have, or is not able to calculate, the probabilities of events occurring. In this case we would classify the situation under the heading of "design under uncertainty". A more detailed distinction between these two classes is made by Weisman /18/.These two approaches to the evaluation of risk are considered separately below.

B1. Design under Risk

This approach has been, most recently, developed for nuclear reactors quite extensively by Otway in a number of papers (see for example /19/). In essence, the approach is to define the risk from a reactor as the sum of all biological risks from all conceivable accidents weighted by the respective accident probabilities. The accident probabilities must be calculated as little or no statistical data is available. The principle means of doing so is through fault tree analysis /20/ which is a means of assessing the likelihood of a complex system performance. The development of a fault tree begins with a definition of the end system fault condition, the undesired event. The system is then methodically analyzed to determine all the logical combinations of functional fault events that can cause the end event. The analysis is wholly dependent on a thorough knowledge of the system functions and configurations. The assumptions involved are:

1. The components of the fault tree are independent
2. The minimal cut sets of the system are known.
   (In everyday language a minimal cut set is the smallest set of primary faults such that if all these primary faults simultaneously exist, than the end system fault condition exists.)

These assumptions are not necessarily always valid. There is no way to guarantee that the tree, when constructed, does not neglect some important mode of failure. For example, the Windscale, SL-1, and NRX accidents all occurred in totally unexpected ways. Furthermore, any tree must necessarily be a minimum fault tree because of these overlooked paths; it therefore can not be conservative unless it is perfect.

Despite the shortcomings of the methods briefly explained above they have been and will be used in the analysis of complex systems, such as nuclear power plants, fuel reprocessing plants, etc. However, the application of the method to the analysis of waste storage facilities is hindered in some respects. The safety of long-term storage, or disposal, facilities has to be guaranteed for hundreds of years. Our

ability to assign probabilities to events that may occur in the distant future is questionable. For example, to list a few of the events that are important for our consideration one would have

1. changes in location of seismic activities
2. changes in location of volcanic activities
3. sabotage and war
4. changes in civilization levels

Of course it may be argued that a catastrophic event that could endanger the waste storage facilities would have more severe consequences than the subsequent release of radioactive materials. In this case the secondary consequences of the event, release of radioactive wastes, are of only passing interest.

B2. Design under Uncertainty

As mentioned above, when we know what conditions or events are possible but cannot estimate the probabilities with which they occur, we are in a condition of uncertainty. It is therefore, difficult to calculate a risk, let alone select an optimum waste management scheme. At the present time, decision theory provides no one best criterion for selecting a strategy under conditions of uncertainty. Instead, there are a number of different criteria. Four of these criteria are discussed below. For this purpose we shall use an example to illustrate the application of the criteria. We shall label the waste management schemes, or strategies, as S_i and the possible events as N_j. In this sense an event would have associated with it a release of radioactive material and subsequently a loss, or burden (X_{ij}) for mankind. Therefore, the matrix for our decision problem would look like the following.

	N_1	N_2	N_3	...	N_m
S_1	X_{11}	X_{12}	X_{13}	...	X_{1m}
S_2	X_{21}	X_{22}	X_{23}	...	X_{2m}
⋮	⋮	⋮	⋮		⋮
S_n	X_{n1}	X_{n2}	X_{n3}	...	X_{nm}

The problem at hand is to select from the n possible strategies, the one most desirable or optimum, through application of some form of criterion. It may be possible, before proceeding, to eliminate one or more strategies from consideration by use of the dominance principle. That is, any strategy S_i would be eliminated from consideration if its losses X_{ij} were greater than the corresponding losses for any other strategy. Mathematically, if $X_{ij} \geq X_{i'j}$ for all j, and further $X_{ij} > X_{i'j'}$ for some j', strategy S_i is dominated by strategy $S_{i'}$, and therefore eliminated. Generally, however, the dominance principle is not itself adequate for making a unique selection from the possible strategies. A unique selection can be made by application of the criterion of pessimism, of optimism, of regret or the subjectivist criterion /21/.

-Criterion of pessimism

Under this criterion the decisionmaker acts in a completely pessimistic manner. It is assumed that Nature would always be malevolent; that is, Nature always acts to maximize our losses. Therefore under this criterion one would select the strategy which minimizes the maximum possible loss. This criterion is sometimes called the "minimax" criterion.

To illustrate simply the application of this criterion assume that our matrix for the waste management strategies is (3x3) with the following fictitious values.

	N_1	N_2	N_3
S_1	5	1	9
S_2	6	3	6
S_3	4	2	7

Decision Matrix

Our fictitious losses could be in units of man-rem's x 10^{-6} (product of the number of people exposed to radiation and the exposure they receive). In this case the worst that could be experienced for each strategy would be

Strategy	Maximum Loss
S_1	9
S_2	6
S_3	7

and we would therefore select strategy S_2 since it minimizes the maximum possible loss.

The argument based on pessimism can be described as a conservative approach to an intrinsically difficult problem. A more refined application of this criterion, as illustrated in the following case, stems from its application to the theory of games /22/.

-Criterion of Optimism

Why should we always assume that Nature will be malevolent? After all, there are times when we get good breaks. Being a complete optimist we would then select strategy S_1 under the assumption that event N_2 would be the determining influence in the future. However, it was not suggested that a rational decisionmaker should be completely optimistic. Rather, a "coefficient of optimism" can be introduced, which takes into account the minimum and maximum

possible losses, by weighting their importance to the decision in accordance with the decisionmakers feeling of optimism. The coefficient of optimism is defined in terms of a standard lottery /22/ between the maximum and minimum losses. In this case it is assumed that the decisionmaker has decided to participate in the lottery since a decision not to entails not managing the waste and therefore, more severe consequences, or losses, for mankind. Therefore,the decisionmaker assigns to the minimum loss a probability which he would be willing to accept in a lottery between the maximum and minimum losses. This probability is then the decisionmakers coefficient of optimism. Assume, for example, that our coefficient of optimism is 0.7. That means that we would be willing to accept the minimum loss if it has a probability of occurance of 0.7 and a maximum loss probability of 0.3. The application of the principle is as follows, assuming that either the maximum or the minimum will occur and with the indicated probabilities.

Strategy	Max. loss	Min. loss	Expected loss
S_1	9	1	$9*0.3 + 1*0.7 = 3.4$
S_2	6	3	$6*0.3 + 3*0.7 = 3.9$
S_3	7	2	$7*0.3 + 2*0.7 = 3.6$

According to this criterion strategy S_1 would be selected since it has the lowest expected loss.

-Criterion of Regret

As suggested by Savage /23/ it is worthwhile to transform our decision matrix before a decision is taken. Savage argues that after a decisionmaker knows the outcome, or event that has occured, he may experience regret because he may wish he had selected a different strategy. Savage therefore suggests that the decisionmaker should attempt to minimize this regret which he can experience.

To illustrate the point, assume that strategy S_2 had been selected and that event N_1 has occurred. The regret experienced would be the difference between 6, the loss experienced, and the lower loss of 4, had strategy S_3 been selected, which in this case would have been better. However, if instead of N_1 event N_3 had occurred, then the decisionmaker would experience no regret since his decision resulted in the lowest possible loss. Following this logic we can construct the following regret matrix.

	N_1	N_2	N_3	Maximum Regret
S_1	1	0	3	3
S_2	2	2	0	2
S_3	0	1	1	1

Regret Matrix

Therefore, to minimize our regret we would select strategy S_3.

- The Subjectivist Criterion

The foregoing illustrative example indicates that decisions under uncertainty depend on which criterion one uses. We would, therefore, like some rational basis for our choice of the criterion. However, it is generally felt that none of the decision criteria under uncertainty satisfies a set of applicable axioms. Each is defective in some respects.

Furthermore, these decision criteria do not use all the information available to the decisionmaker. For example, "minimax" considers only the worst X_{ij} for each N_j; all the other outcomes are ignored. Rationality demands that the decisionmaker, in making his decision, should consider the effects of all the information available to him. This can be accomplished by the extension of the subjectivist criterion explained below.

The subjectivist would maintain that the decisionmaker has useful information in the form of degrees of belief concerning the likelihoods of occurrence of the relevant states of nature. The subjectivist would consider our problem as any other decision problem under risk.

The criterion recommended by the subjectivists is called the Laplace criterion. Under this criterion if the probabilities with which the various events will occur are not known, we assume that they are all equal. We then calculate the expected loss for each strategy and select that strategy which has the lowest expected loss. In our case we have three possible states of nature and we would therefore assume that each would occur with a probability of 1/3.

Strategy	Expected loss
S_1	1/3x(5+1+9) = 5.0
S_2	1/3x(6+3+6) = 5.0
S_3	1/3x(4+2+7) = 4.33

Therefore, according to the Laplace criterion we would select strategy S_3.

Normally, however, the decisionmaker has some feeling about the likelihood of occurrence of the various events. He could as well use his, or a combination of other's judgmental probabilities to resolve his decision problem. The obtaining and use of judgmental probabilities is quite clearly explained and advocated by Raiffa /24/. In this respect we can use the technique of "sensitivity probing" to determine how unbalanced the probabilities would have to be to obtain a shift in the selection of a strategy. It may be that the selection of a strategy is not very sensitive to the probabilities, the unbalance of which are within acceptable range based on judgment of the decisionmaker, or a consensus of experts.

C. Specific Problems Related to Risk Evaluation of Radioactive Waste Management

There are problems inherent in the risk evaluation of radioactive waste management schemes that complicate the evaluation considerably. Present available methods for risk evaluation are not readily applicable here due to many reasons. The first is the addition of the time dimension to the problem, and the closely related problem of one being almost totally incapable of calculating event probabilities in the time dimension of periods longer than that of recorded history. These problems are discussed in the following.

The nuclear fuel cycle has associated with it two types, or elements, of risk. The first is the risk due to the operation of the facilities that compose the fuel cycle. In this argument the final waste disposal facility is not considered in operation after final closure. This risk to health, or death, is a result of possible accident conditions of the facility during operation and of the normal operational releases of radioactive materials. The second element of risk is due to the generation of the radioactive wastes. The radioactive wastes released to the environment immediately, or after a delay time, in the liquid and gaseous effluents of the facilities are considered a part of the first type of risk. The major portion of the waste, in terms of radioactivity, is however contained and handled in a manner respecting its potential biological hazard. The main distinction between these two risk elements is that the first does not have a time factor in it. In other words, the first risk element can be effectively "shut-off" by simply discountinuing operation of the nuclear fuel cycle facilities. The second element is, more or less, a permanent risk which will be inherited by future generations, i.e., if we have no means of removing all the wastes permanently from our environment. We do not presently possess a means of eliminating the risk due to the radioactive waste even though such means are under intensive investigation. These two elements of risk, as we call them, are illustrated graphically in Fig. 7. In this figure

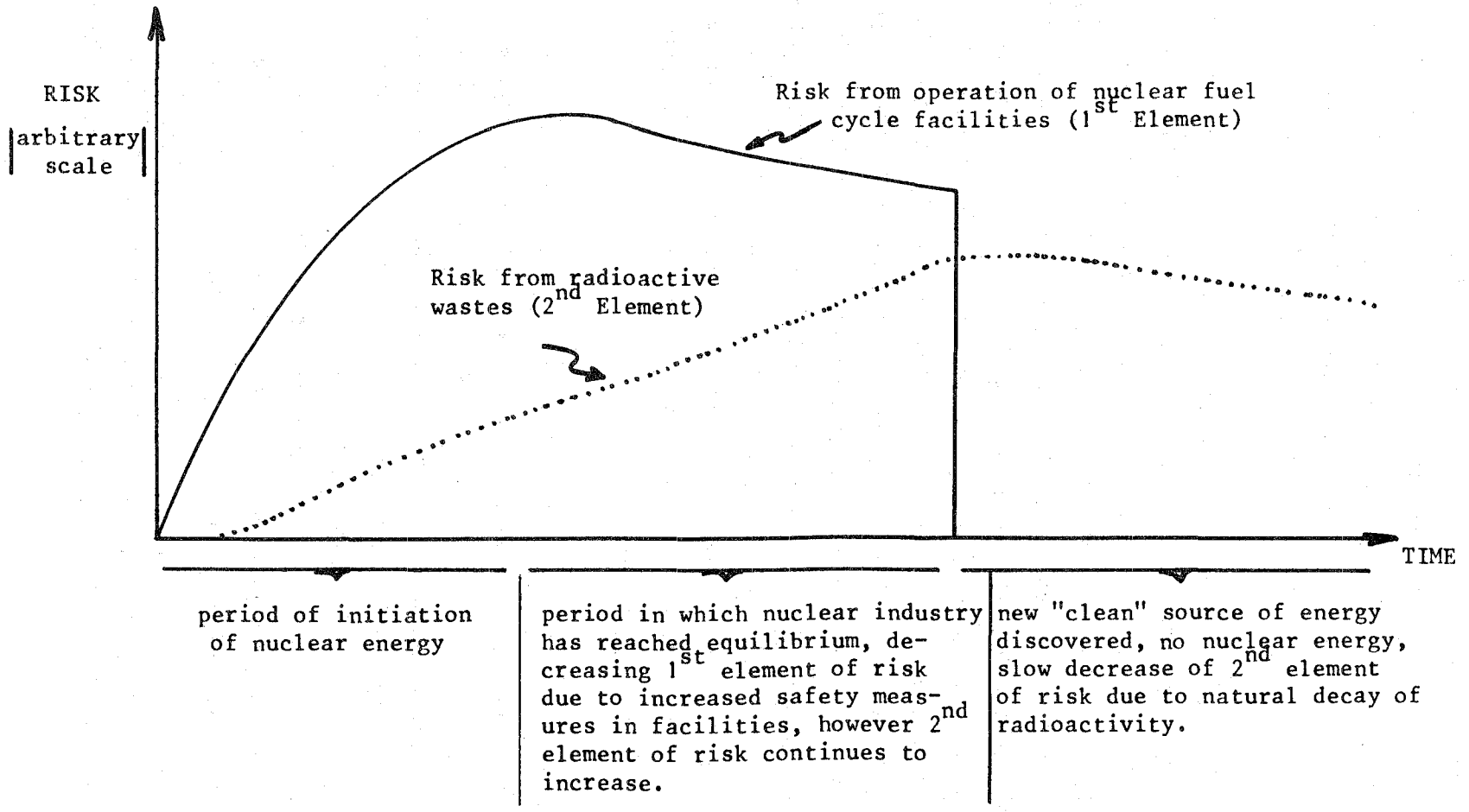


Fig. 7 Elements of Risk Associated with the Nuclear Fuel Cycle

no attempt has been made to maintain a realistic scale on the coordinates or a reasonable ratio between the two curves. It must be mentioned that there is a slight overlap of the elements of risk. For instance, some long half-life isotopes, e.g. ^{129}I, are released in the gaseous effluents of the facilities and are reconcentrated by organisms. Once this isotope is released it represents an irreversible commitment to the environment and a source of long-term risk.

Notice in the figure that after discontinued operation of the nuclear fuel cycle the risk due to the radioactive wastes decrease slowly with time. This would be expected due to the natural decay of the waste isotopes to stable isotopes. The time period for the curve to reach an effective "zero point" is, however, very long, literally millions of years. It is, however, possible that rather than decrease, the curve could rise for several possible reasons. One obvious reason is a decline in the technological capabilities of our civilization. Any reader of the book "The Limits to Growth" can appreciate this; civilization in the future may not have the material means of handling possible "breakdowns" in our waste disposal facilities. A second possible reason for an increase is an unintentional "opening" of a waste disposal facility. A third possibility would be a catastrophic breakdown in the waste facilities due to presently unforeseen circumstances which could not be handled even with an advanced technology.

Another important point in this respect is a definition of an effective "zero point" for the second element of risk. There is obviously a risk associated with the wastes until the time the last atom has decayed even though the risk due to one atom would be considered negligible. The time the last atom would decay is infinity. Therefore, it is desirable to define a point in time after which we would consider the wastes to be no longer "dangerous". A convenient means of defining the "zero point" would be by comparison of the waste disposal facility to naturally occurring risks of the same type. For example, the zero point would be the time when inadvertent opening of the waste facility is no more hazardous than that due to exposing uranium or thorium ore. A time period so defined may, indeed, be very long.

A second approach, and also a supplement to the above concept, is that of a "social rate of discount". The concept involved is that we may intuitively equate the value of death, or well-being, of a person today with that of $(1.0 + i)^n$ persons sometime in the future. If n is an integer and expresses the number of years hence, than i would be the social rate of discount. To put it another way, the risk of one life today to secure the safety of one life in the near future has no meaning. However, to secure the safety of 100 lives in the near future does have meaning to us.

If one is able to obtain a rate of discount in this respect then it would be possible to find the upper limit of additional risk one is willing to accept today from the waste management schemes to secure an additional degree of safety, say in 300 years from now. This would be possible even with only an order of magnitude estimate of the number of people that would be endangered by a release of material from the disposal facility in 300 years. Two possible disposal extremes would be total radioactive waste disposal by deep space disposal or by hydraulic-fracturing slightly below the earth's surface. The first scheme is probably the most expensive in cost and safety for the present generation, but can guarantee the highest degree of safety for future generations. The second scheme is very inexpensive in cost and safety for us today, but probably the most expensive in safety in the long run. Someplace between the two alternatives lays an optimum which can be evaluated with the concept of social rate of discount. It may turn out that the natural decay of the radioactive waste would more than counterbalance the effect of time-dependent concept of risk with the social rate of discount concept, depending of course on the magnitude of the rate of discount.

Very few attempts at obtaining a social rate of discount have been made. One such attempt was made by Feldstein /25/ to measure society's marginal rate of substitution between consumption in consecutive years. However, what we seek is a measure of what we are willing to sacrifice today for the exclusive benefit of others that will live after us. A possible measure of this is life

insurance. Life insurance is something we purchase (sacrifice a portion of our disposable income) to guarantee the level of well-being of people whose present well-being is the purchaser's responsibility. However, with present available data on life insurance (see for example the Life Insurance Fact Book /26/) it may not be possible to evaluate the desired social rate of discount.

The preceding remarks illustrate the difficulty of applying so-called "standard" risk analysis methods to the evaluation of risk from waste management schemes, or even for the purpose of selecting an optimum waste management strategy. For example, in the decision matrix of the preceding section the elements of the matrix are not mutually exclusive since the period of time over which events can occur, and be of interest for the decision process, is very long. In addition, many of the elements of the matrix will not in any way represent a "loss" to the decisionmaker himself. Nevertheless, it is felt that these points can be resolved by the application of a technique being developed in Operations Research. The application of this technique, called utility theory, is illustrated by use in two widely different areas, the first by Turban and Metersky /27/, and the other by Keeney /28/. Inherent in the application of utility theory is the use of judgmental probabilities and involves a procedure somewhat similar to that explained under the subsection "The Subjectivist Criterion" in section III.B2.

IV. Hazard Index of Radioactive Wastes

Before one can consider any disposal system for any type of hazardous material it is absolutely necessary to know the characteristics of the material. In addition, the time behaviour of the properties of the material and the relative ecological importance of the various components of the material (if there are more than one) must be known. Therefore, before we discuss waste disposal schemes for radioactive wastes we examine the characteristics of these wastes.

Typically, radioactive wastes are characterized by curies (disintegrations/sec.) per unit volume of the waste as a function of the time from the date that the reactor fuel was reprocessed. This type of characterization is not ideal since the biological effect of the waste is only partially determined from the curies. More important is the type of decay particle and its energy, which are inherent properties of the decaying isotope. In addition, it is important to know the relative ease with which the isotopes move through our environment, the extent to which they are reconcentrated by organisms directly, or indirectly, involved in man's food chain, and our own critical organ's characteristic rate of accumulation and elimination. Therefore,to properly design a waste disposal system it is necessary to have an index for each radioactive isotope which reflects all of these properties. The establishment of such a true index is a very difficult problem and can only be partially accomplished today due to the lack of sufficient data.

As a first approximation to a hazard index we have used the reciprocal of the established maximum permissible concentration of the isotope in water, MPC_w (Ci/m^3). In other words, the hazard index (HI^i) of Q^i curies of isotope i in a radioactive waste mixture is defined as

$$HI^i = Q^i/MPC_w^i. \qquad (1)$$

The units of the hazard index are volume (m^3) of water required to dilute the radioactive isotope to acceptable limits. The total hazard index of the waste is found by summing over all isotopes present;

$$HI = \sum_i HI^i = \sum_i \frac{Q^i}{MPC_w^i}. \qquad (2)$$

It has to be emphasized that even though we use this hazard index throughout this work, there are some severe shortcomings involved in the definition.

The basic procedure in setting the maximum permissible concentrations for the general public is to calculate that amount of each radioisotope in water or air which after 50 years of continuous intake or inhalation would result in a dose to the critical body organ equal to the maximum permissible dose. In order to ensure uniformity it has become customary to compute the MPC's for a "standard man" (see Ref. /29/). The actual dose which an individual will receive from the maximum permissible body burden will thus depend on whether he is larger or smaller than the standard man and whether his relevant physiological processes are on the high or low side of normal.

The criteria for chronic exposure of the public should, however, be related to maximum permissible doses rather than to MPC's since the latter do not explicitly consider perhaps more limiting pathways /30/ of exposure than those caused by inhalation of air or ingestion of water. Special considerations should also be given to the specific location of the facilities treating the wastes as there may be mechanisms available for the reconcentration of the radioactive isotopes and pathways for ingestion by the public. Such pathways are illustrated in the two diagrams of Fig. 8 and 9.

Finally, the hazard of a radioactive noble gas is represented better by the MPC for inhalation than the MPC for ingestion. In addition, the other isotopes can be considered to have a residence time in the atmosphere as well as on the ground. That is, it is entirely possible that the isotopes can be released to the atmosphere, as a result of an accident or normal suspension mechanisms, deposited on the ground only to be later resuspended. Therefore,the contribution to the hazard from these two exposure mechanisms could be obtained as a weighted sum. Since we cannot in any case, add meaningfully volumes of air and water we convert the MPC's to maximum permissible annual intake, MPI (Ci/yr). The MPI's are obtained by multiplying the respective MPC by the volume of water ingested (0.8 m^3) or the volume of air inhaled (7300 m^3) annually by the "standard man". Our hazard index would be in this case for isotope i

$$HI^i = Q^i \left[\frac{a^i}{MPI_w^i} + \frac{b^i}{MPI_a^i} \right] \qquad (3)$$

where MPI_a^i = maximum permissible annual inhalation of isotope i (Ci/yr.)

and a^i, b^i are appropriate weighting factors for isotope i.

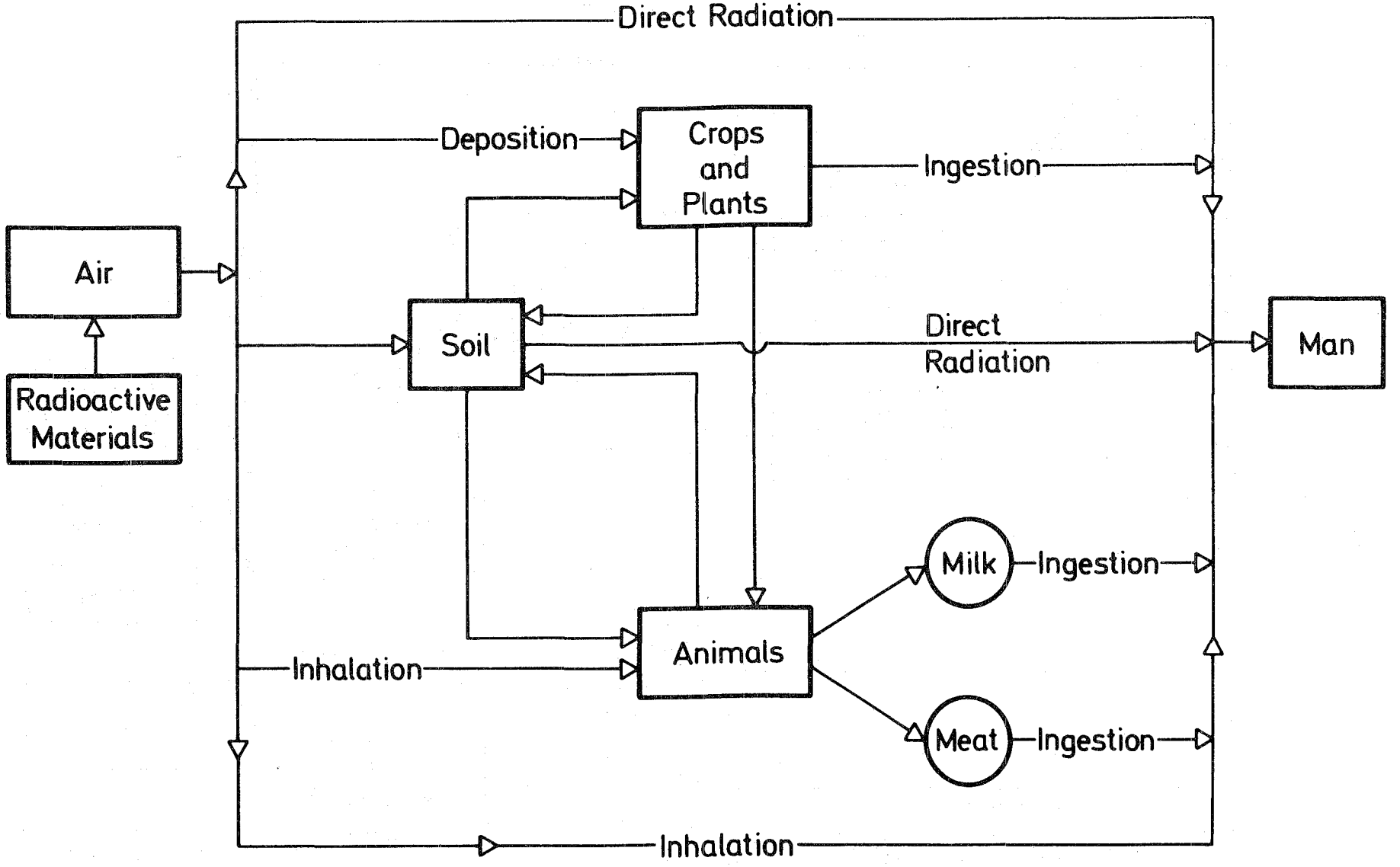


Fig.8 Pathways between Radioactive Materials released to the Atmosphere and Man (Environmental Protection Agency, 1972)

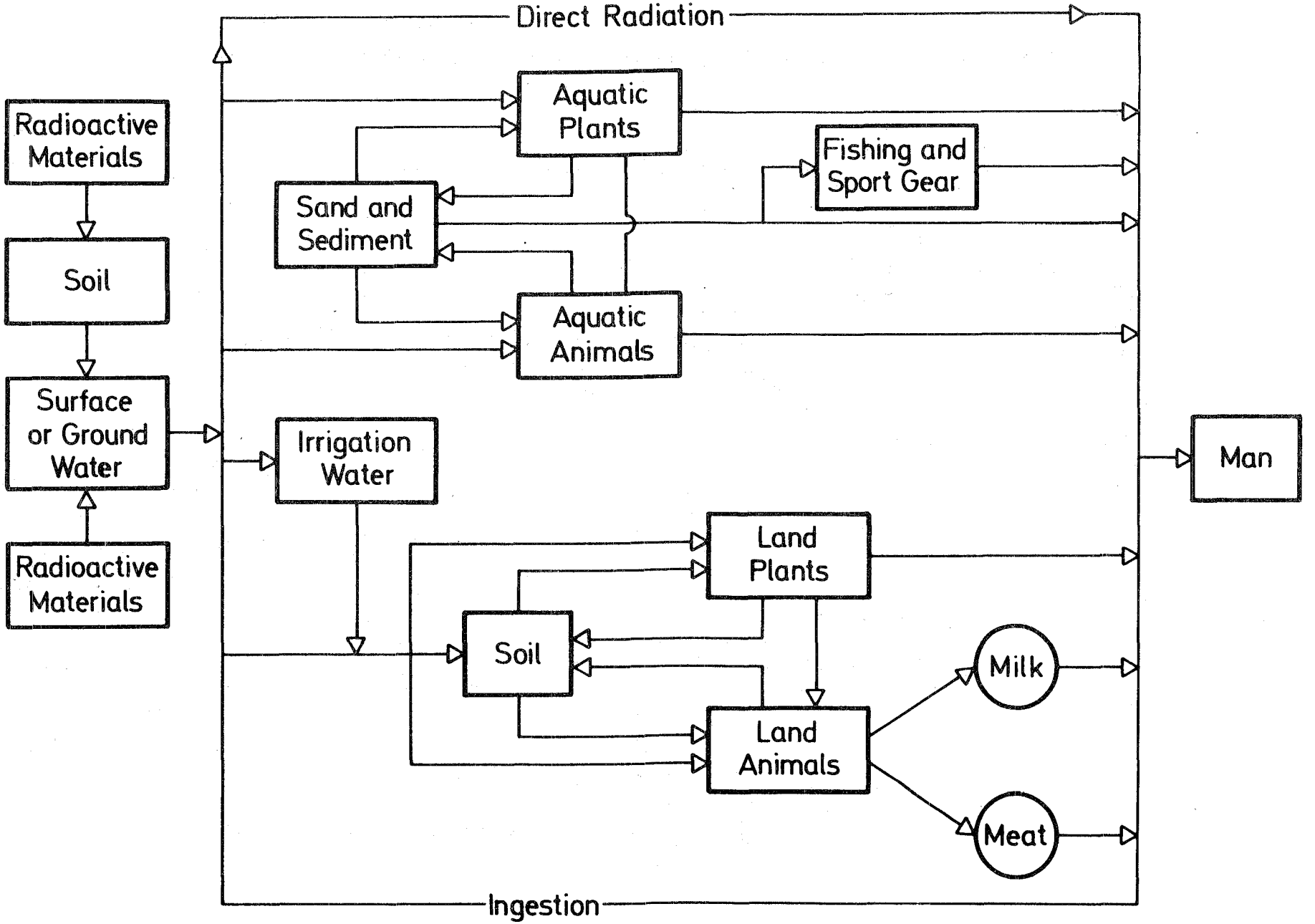


Fig.9 Pathways between Radioactive Materials released to Ground and Surface Waters and Man

An attempt to define a hazard index has also been made by Gera and Jacobs /31/. They choose to define a potential hazard index (PHI) for an isotope i as

$$PHI^i = P_i * \frac{Q^i}{MPI^i} * \frac{T^i}{0.693} \qquad (4)$$

where

Q^i = total activity of isotope i (Ci)

MPI^i = maximum permissible annual ingestion of isotope i (Ci), and

T^i = physical half-life of isotope i (years).

P_i is a factor dependent on the biological availability of radioisotope i once it is dispersed into the environment and on the reliability of waste containment, and represents the probability of the nuclide leaving the site of disposal or handling, and reaching man. Because of the limited data available they set P_i equal to one. The inclusion of the mean life ($T^i/0.693$) in the definition of the hazard index PHI represents a measure of the time span during which the radioisotope will exist. Gera and Jacobs consider also a weighting of the hazard index for inhalation and ingestion of the form shown in Eq. (3).

Comparing the hazard index used in this work, Eq. (1), and that in Eq. (4) one can make the following important remark, realizing that

$$MPC_w^i * 0.8 = MPI_w^i.$$

If P_i is set equal to one then the difference between Eq. (1) and Eq. (4) is that Eq. (1) represents an instantaneous hazard, or risk, at the time one has Q^i curies of isotope i. On the other hand, Eq. (4) represents a time integrated hazard, or risk, from the point in time one has Q^i curies of isotope i to infinity. For example, by integrating the exponential equation for radioactive decay,

$$Q^i(t) = Q_o^i e^{-\lambda_i t},$$

where Q_o^i is the curies present at t = o , one finds

$$\int_0^\infty Q^i(t)dt = Q_o^i \int_0^\infty e^{-\lambda_i t}\, dt = \frac{Q_o^i}{\lambda_i} = Q_o^i \frac{T^i}{0.693} .$$

Therefore, if one calculates the values of Eq. (4) as a function of the age of the waste one has, in effect, at each time point the hazard integrated forward in time. The use of Eq. (4) implies, therefore, an assumption involving the social rate of discount of risk; that is, there is no rate of discount used. This does not represent, however, human intuitive thinking as discussed in section III. It is for this reason that we choose to use the hazard index defined by Eq. (1).

The inclusion of a social rate of discount in the Eq. (4) is quite easily performed by assuming an exponential function for the value of future risks as perceived today with a discount rate of α/year; i.e. $e^{-\alpha t}$. Introducing this into the integration of exponential decay equation one finds

$$\int_0^\infty Q^i(t)e^{-\alpha t}\, dt = Q_o^i \frac{T^i}{0.693 + \alpha T^i}$$

and therefore Eq. (4) becomes, with $P_i = 1.0$,

$$PHI^i = \frac{Q^i}{MPI^i} \times \frac{T^i}{0.693 + \alpha T^i} \qquad (5)$$

Setting $\alpha = o$ one obtains Eq. (4).

As mentioned above Eq. (1) does, however, suffer from some limitations. These limitations could be relaxed by the inclusion of a factor P_i, as

defined in Eq. (4), and the appropriate weighting factors defined in Eq. (3), if sufficient data were available. The calculation of these factors would depend on such parameters as the form of the waste composition, the location of the waste treatment and storage facility, etc. For the general discussion that follows in the next section it is sufficient, to a first approximation, to use Eq. (1) for the hazard index. From this hazard index it is possible to get a relative idea of the importance of the various radioactive isotopes present in the wastes and the possible gains, or losses, to be achieved through different waste handling procedures.

The hazard index of Eq. (1) has been calculated for the spent fuel composition of seven different cases for a time span of 1 to 10^6 years after reprocessing of the fuel. In all cases it was assumed that reprocessing of the fuel occurs 150 days after discharge from the reactor. The plutonium and uranium losses to the waste stream were assumed to be 1.0% of the quantity present in the spent fuel. In addition, the same loss was assumed for thorium from the high temperature reactor fuel. These losses are representative of today's technology. All other heavy metals present in the fuel are assumed to follow the waste stream. The cases calculated are the following:

1) PWR /13/ with an equilibrium uranium fuel cycle,
   3.3% initial enrichment
   34 000 MWD/T burnup

2) PWR with plutonium recycle fuel ($1^{\underline{st}}$ recycle)
   ≈19% of fissile charge plutonium
   34 000 MWD/T burnup

3) same as 2) except $2^{\underline{nd}}$ recycle of plutonium

4) LMFBR /4/ fueled with plutonium from an LWR reactor industry, core and blanket are mixed proportionally in reprocessing, 34 000 MWD/T burnup of mixture

5) same as 4) except LMFBR fueled with fast breeder equilibrium plutonium

6) THTR /15/ high temperature reactor reference design
   ^{233}U recycle fuel

7) HTGR same as 6) except no ^{233}U recycling equilibrium uranium fuel of 93% enrichment.

The burnup calculations were performed as explained in the previous section II.C3. In the calculation 461 fission product isotopes were considered, as well as all heavy metals from ^{207}Tl to ^{253}Es. Activation products of the fuel element cladding, element spacers, etc. were also calculated.

The results of these calculations per ton fuel (1000 kg) are given in Figs. 10 to 22. No losses are assumed for any of the fission products, as for example, ^{85}Kr and ^{3}H which do not follow the waste stream. The curves are, therefore, not meant to represent a particular waste composition but rather to illustrate the relative importance of the individual isotopes present. In the heavy metal curves, for example Fig. 11, the individual elements labeled refer to the element and its daughter products.

At this point one can make some interesting discoveries from these curves.

From the fission product curves one notices that the isotopes fit into two distinct classes, those that essentially vanish within 1000 years and those that are practically constant over the 10^6 year time span. The two isotopes that determine the envelope are ^{90}Sr and ^{129}I. The fission products determine the total waste hazards up to about 600 years, after which time the heavy metals dominate. The difference in the hazards of the fission products and heavy metals from 600 years to 10^6 years is only 1 to 2 orders of magnitude.

As points of reference the hazard index for fresh fuel compositions, i.e. for the fuel before it is placed in the reactor, have also been calculated. These are the following per ton fuel:

HI (PWR case 1) = 6.49×10^4

HI (PWR case 2) = 1.66×10^9

HI (LMFBR case 4) = 1.16×10^{10}

HI (LMFBR case 5) = 2.49×10^9

HI (THTR case 6) = 9.67×10^6

HI (HTGR case 7) = 5.52×10^4

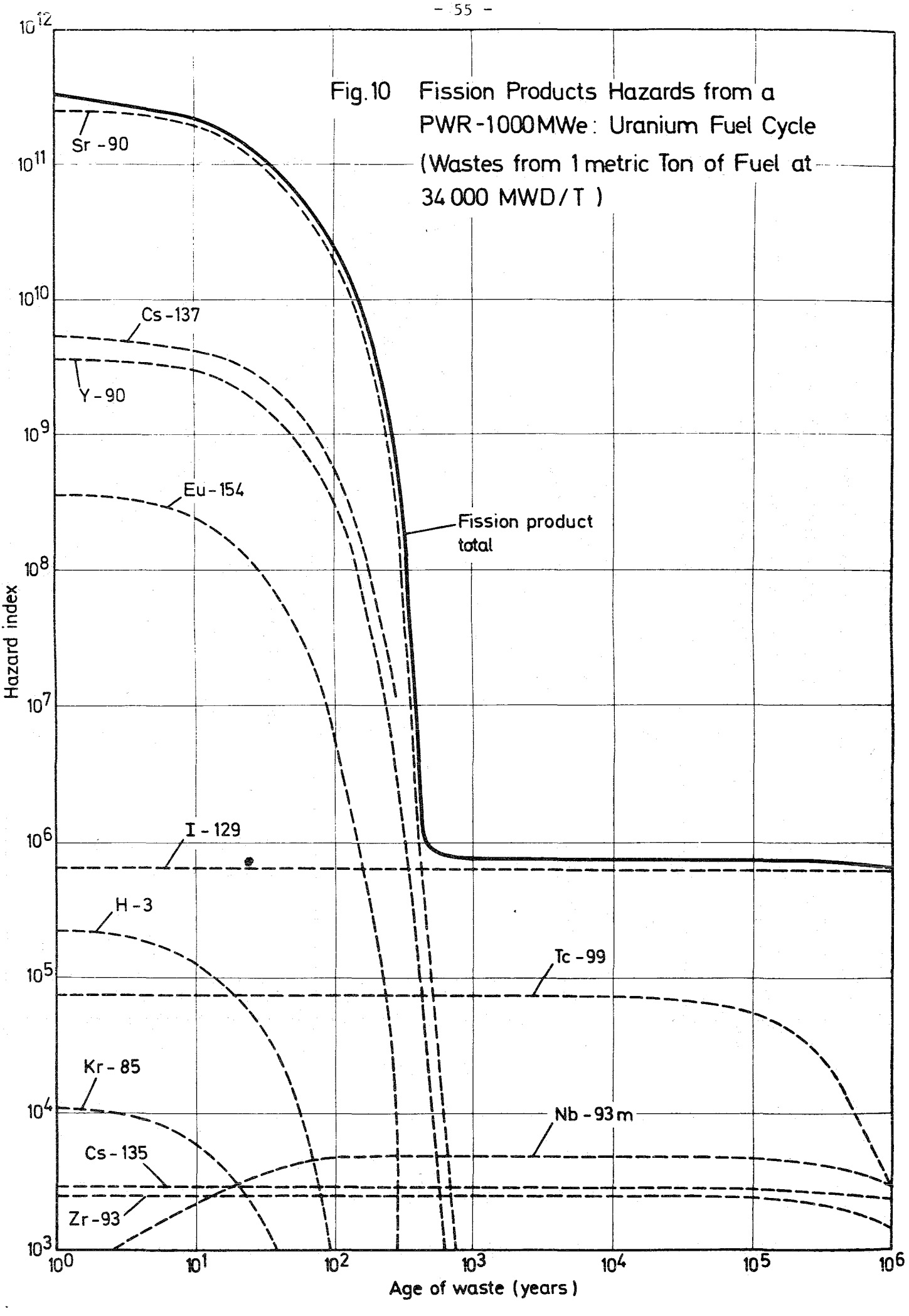


Fig. 10 Fission Products Hazards from a PWR-1000MWe: Uranium Fuel Cycle (Wastes from 1 metric Ton of Fuel at 34 000 MWD/T)
Sr-90
Cs-137
Y-90
Eu-154
Fission product total
I-129
H-3
Tc-99
Kr-85
Nb-93m
Cs-135
Zr-93
Hazard index
Age of waste (years)
10^{12}
10^{11}
10^{10}
10^{9}
10^{8}
10^{7}
10^{6}
10^{5}
10^{4}
10^{3}
10^{0}
10^{1}
10^{2}
10^{3}
10^{4}
10^{5}
10^{6}

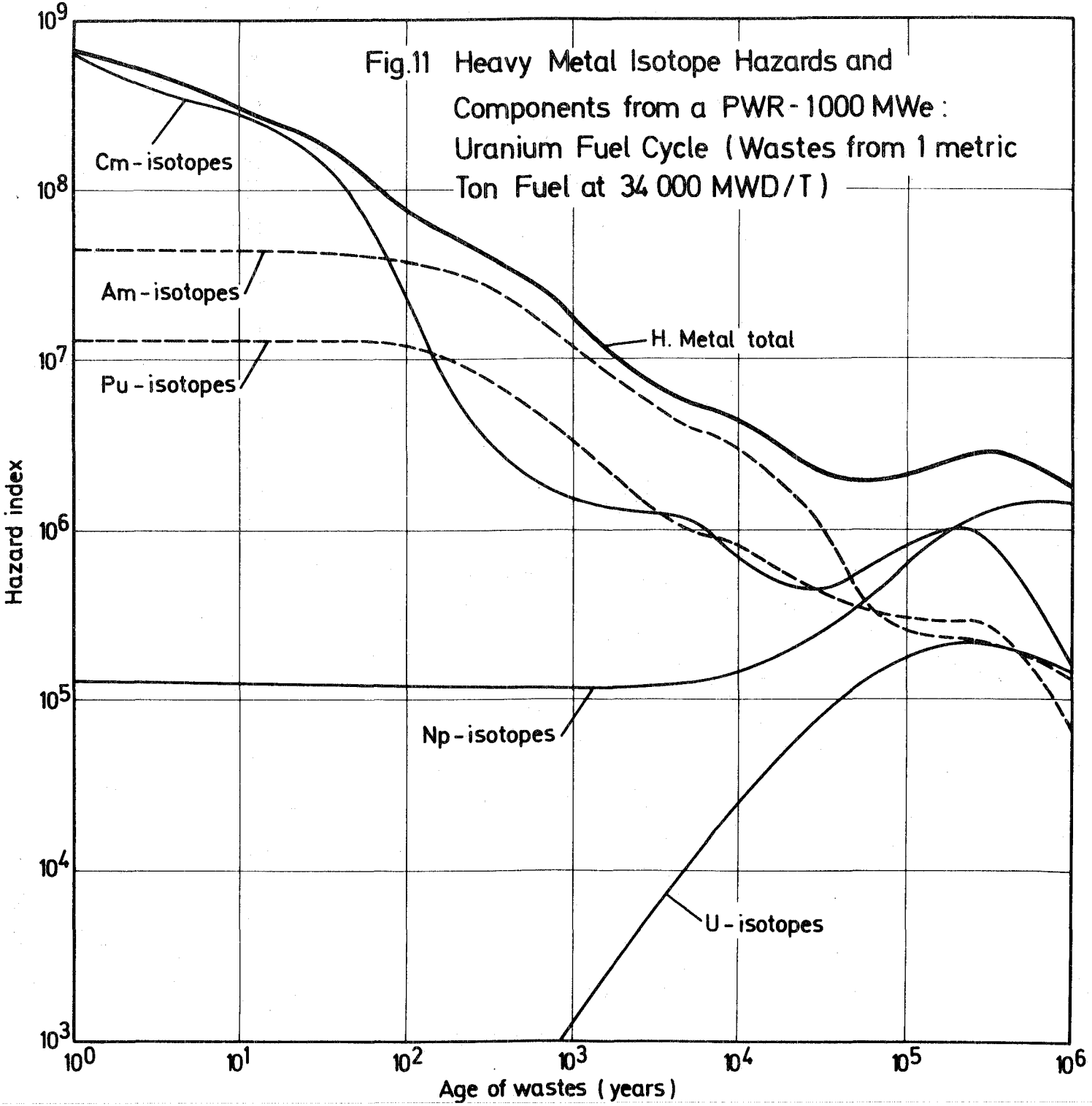


Fig.11 Heavy Metal Isotope Hazards and Components from a PWR - 1000 MWe : Uranium Fuel Cycle (Wastes from 1 metric Ton Fuel at 34 000 MWD/T)
Cm - isotopes
Am - isotopes
Pu - isotopes
H. Metal total
Np - isotopes
U - isotopes
Hazard index
Age of wastes (years)
10^9
10^8
10^7
10^6
10^5
10^4
10^3
10^0
10^1
10^2
10^3
10^4
10^5
10^6

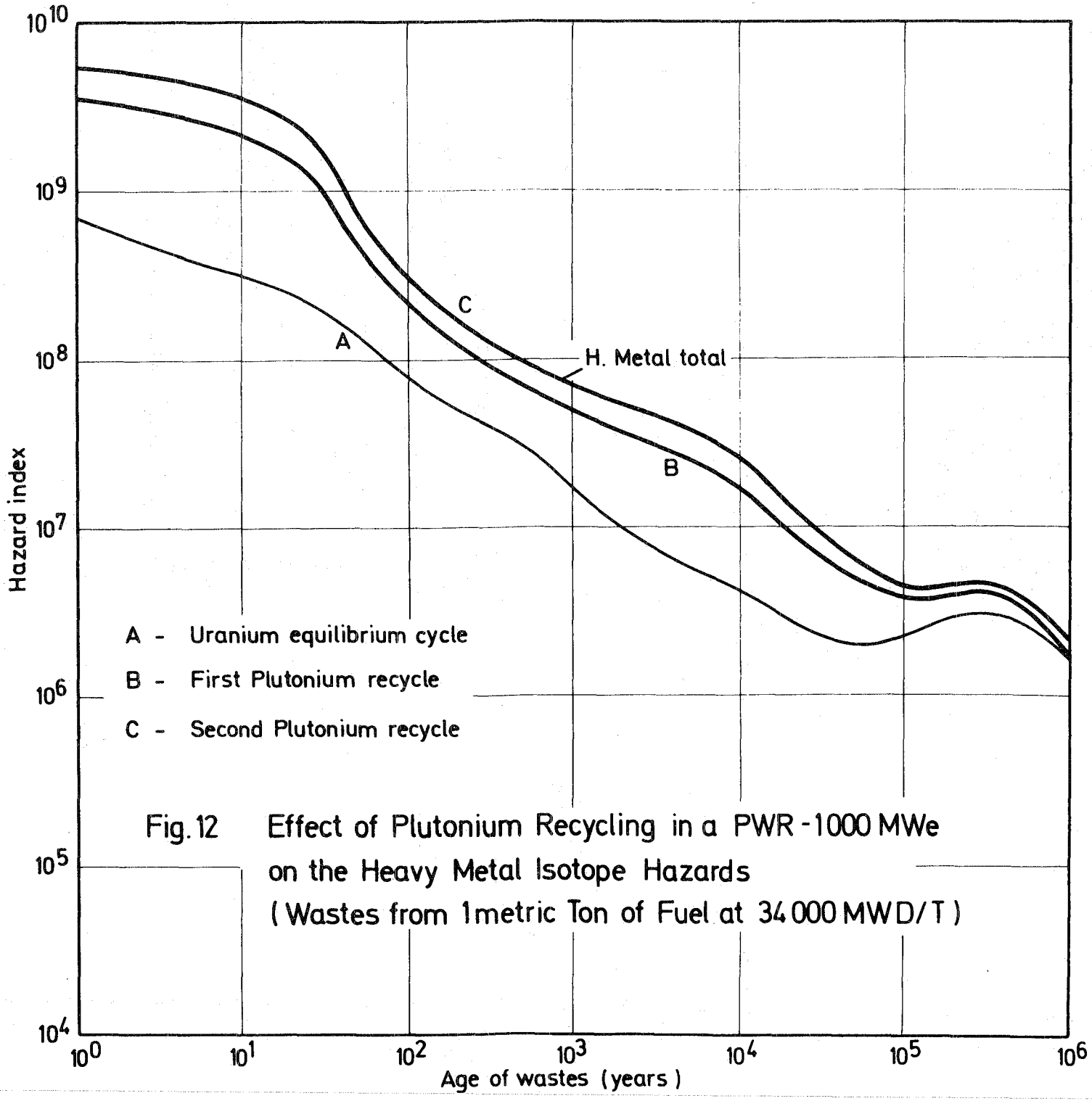


Fig. 12 Effect of Plutonium Recycling in a PWR - 1000 MWe on the Heavy Metal Isotope Hazards (Wastes from 1 metric Ton of Fuel at 34 000 MWD/T)

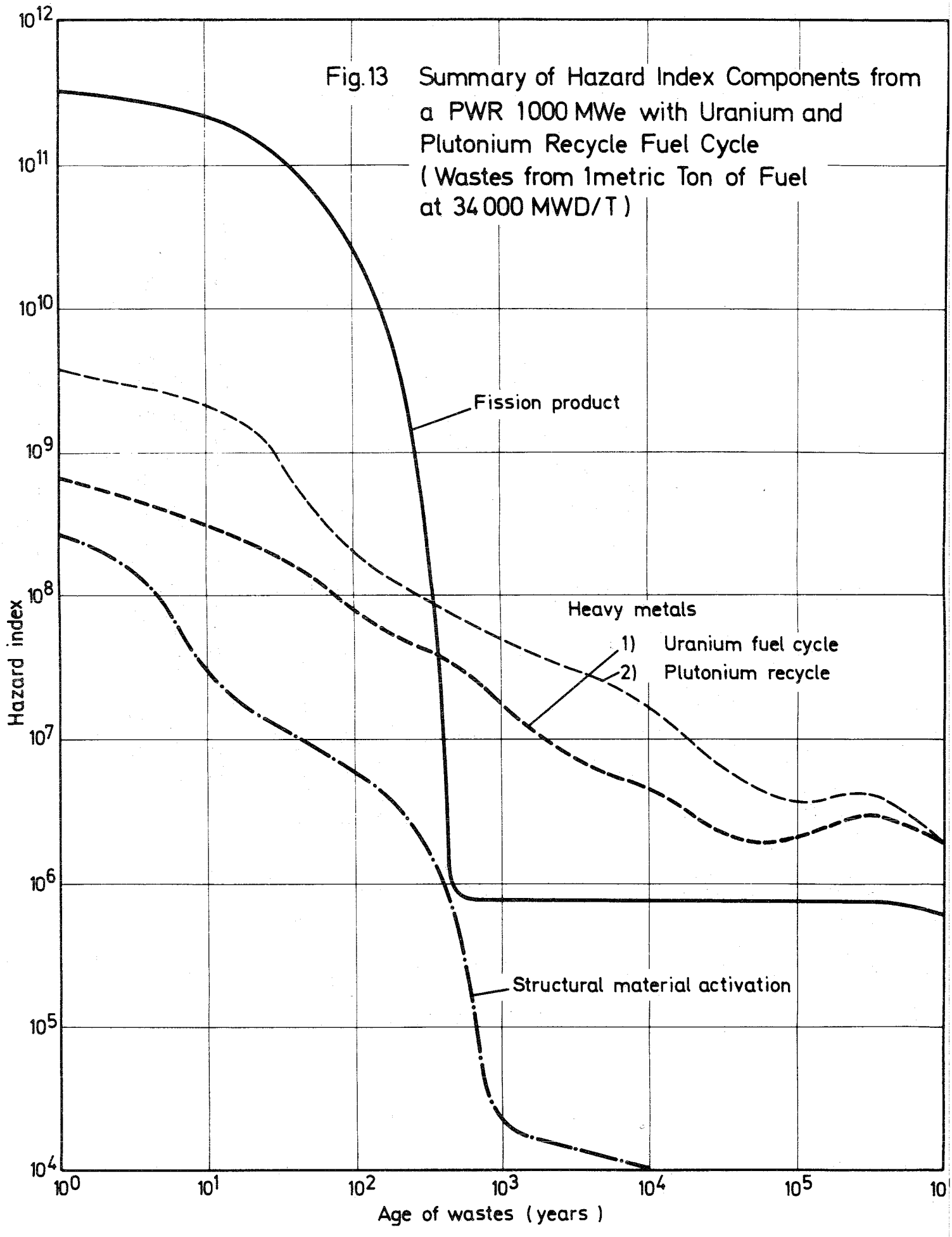


Fig.13 Summary of Hazard Index Components from a PWR 1000 MWe with Uranium and Plutonium Recycle Fuel Cycle (Wastes from 1metric Ton of Fuel at 34 000 MWD/T)
Fission product
Heavy metals
1) Uranium fuel cycle
2) Plutonium recycle
Structural material activation
Hazard index
Age of wastes (years)
10^{12}
10^{11}
10^{10}
10^{9}
10^{8}
10^{7}
10^{6}
10^{5}
10^{4}
10^{0}
10^{1}
10^{2}
10^{3}
10^{4}
10^{5}

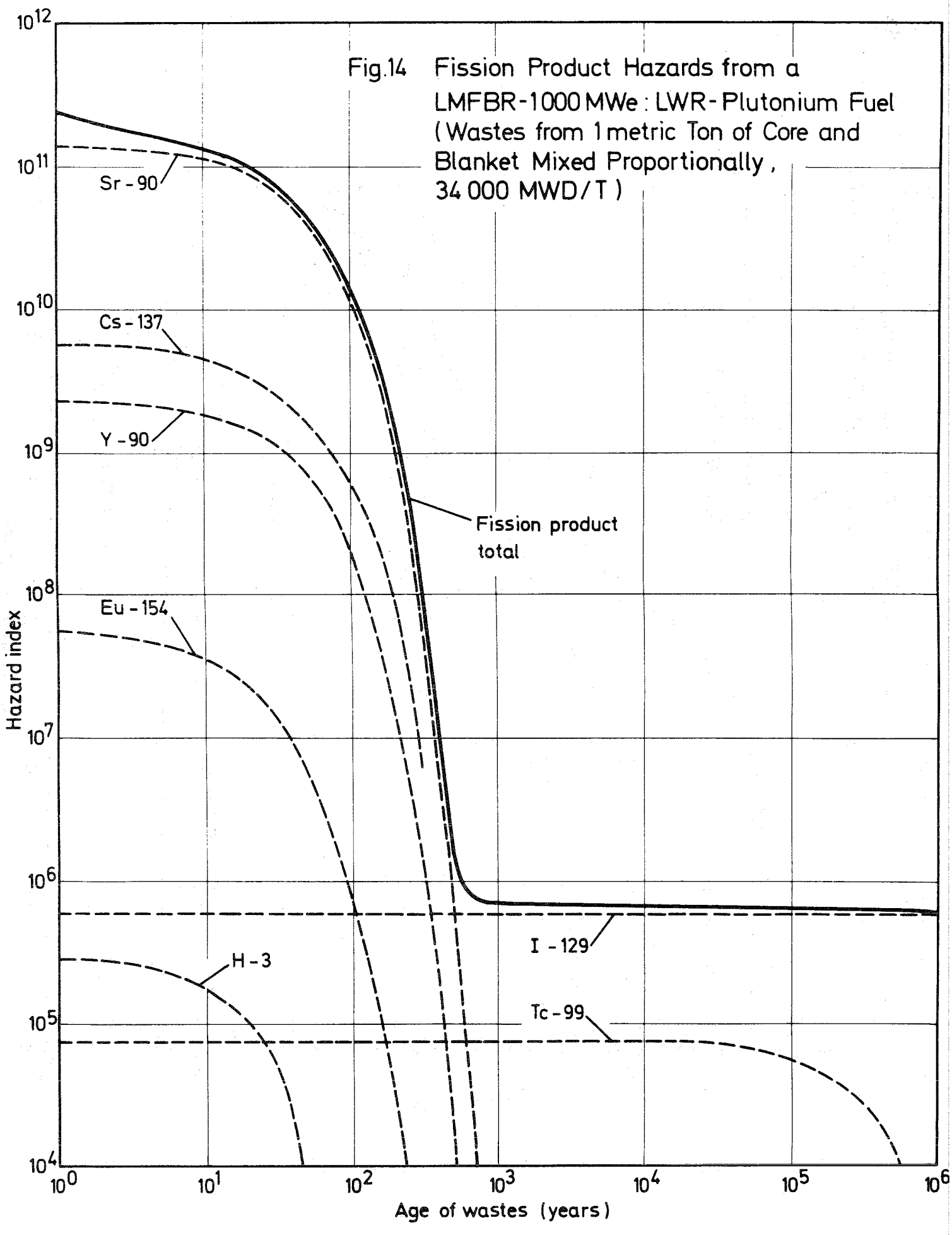


Fig.14 Fission Product Hazards from a LMFBR-1000 MWe : LWR-Plutonium Fuel (Wastes from 1 metric Ton of Core and Blanket Mixed Proportionally, 34 000 MWD/T)
Sr-90
Cs-137
Y-90
Fission product total
Eu-154
I-129
H-3
Tc-99
Hazard index
Age of wastes (years)
10^{12}
10^{11}
10^{10}
10^{9}
10^{8}
10^{7}
10^{6}
10^{5}
10^{4}
10^{0}
10^{1}
10^{2}
10^{3}
10^{4}
10^{5}
10^{6}

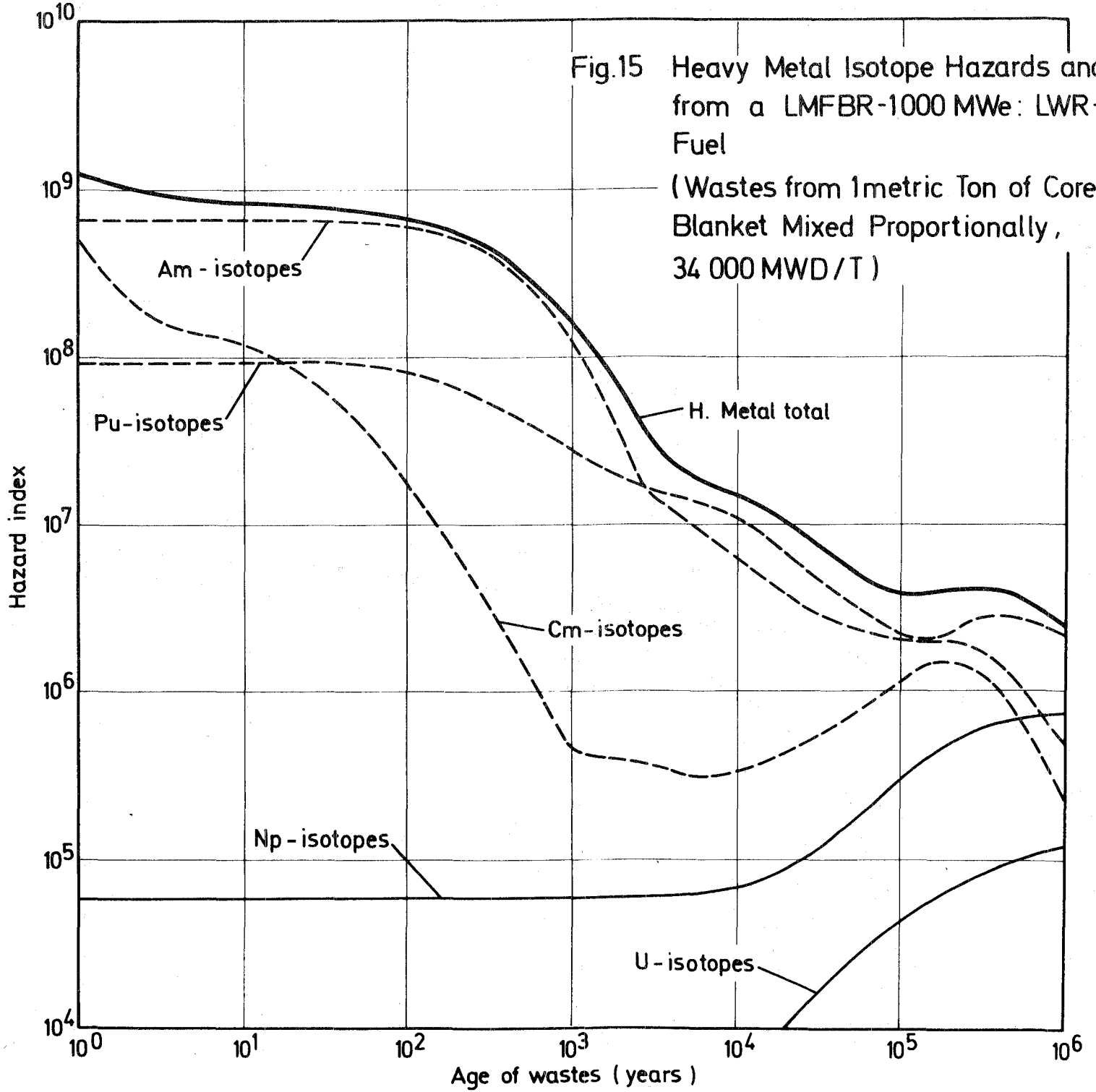


Fig.15 Heavy Metal Isotope Hazards and Components from a LMFBR-1000 MWe: LWR-Plutonium Fuel
(Wastes from 1metric Ton of Core and Blanket Mixed Proportionally, 34 000 MWD/T)

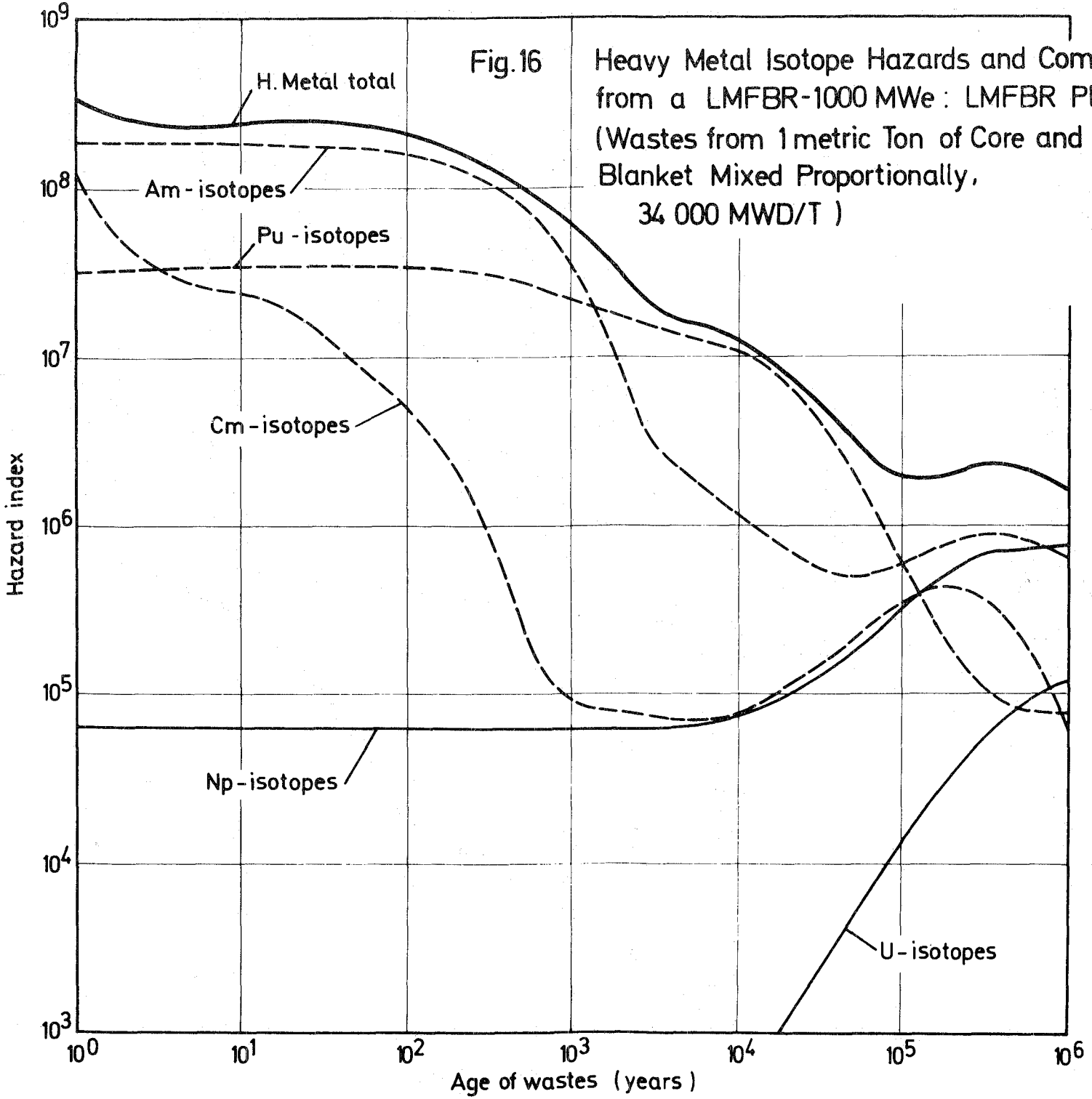


Fig. 16
Heavy Metal Isotope Hazards and Components from a LMFBR-1000 MWe : LMFBR Plutonium Fuel (Wastes from 1 metric Ton of Core and Blanket Mixed Proportionally, 34 000 MWD/T)
H. Metal total
Am - isotopes
Pu - isotopes
Cm - isotopes
Np - isotopes
U - isotopes
Hazard index
Age of wastes (years)
10^9
10^8
10^7
10^6
10^5
10^4
10^3
10^0
10^1
10^2
10^3
10^4
10^5
10^6

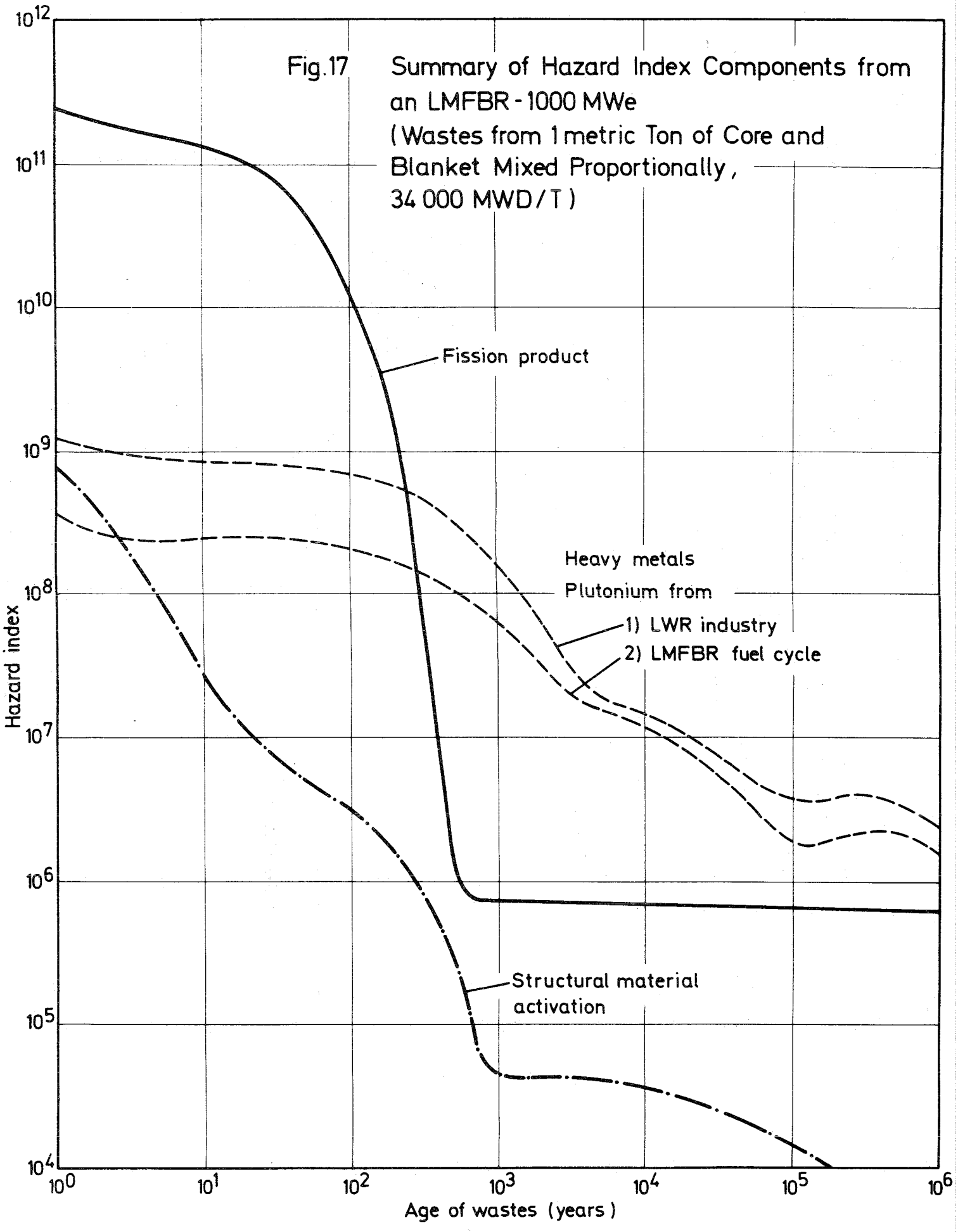


Fig.17 Summary of Hazard Index Components from an LMFBR - 1000 MWe (Wastes from 1 metric Ton of Core and Blanket Mixed Proportionally, 34 000 MWD/T)
Fission product
Heavy metals
Plutonium from
1) LWR industry
2) LMFBR fuel cycle
Structural material activation
Hazard index
Age of wastes (years)
10^{12}
10^{11}
10^{10}
10^{9}
10^{8}
10^{7}
10^{6}
10^{5}
10^{4}
10^{0}
10^{1}
10^{2}
10^{3}
10^{4}
10^{5}
10^{6}

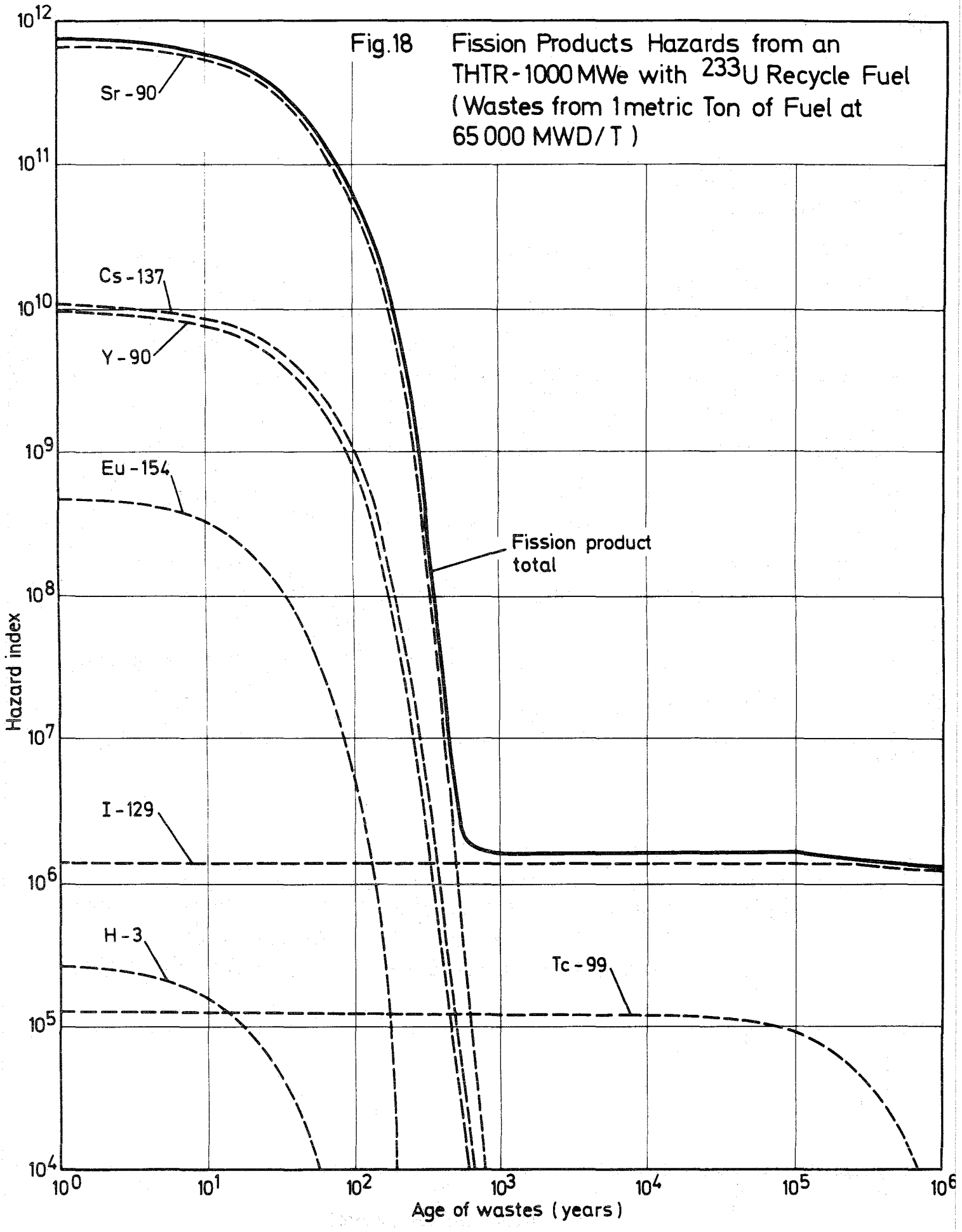


Fig.18 Fission Products Hazards from an THTR-1000 MWe with 233U Recycle Fuel (Wastes from 1 metric Ton of Fuel at 65 000 MWD/T)
Sr-90
Cs-137
Y-90
Eu-154
Fission product total
I-129
H-3
Tc-99
Hazard index
Age of wastes (years)
10^12
10^11
10^10
10^9
10^8
10^7
10^6
10^5
10^4
10^0
10^1
10^2
10^3
10^4
10^5
10^6

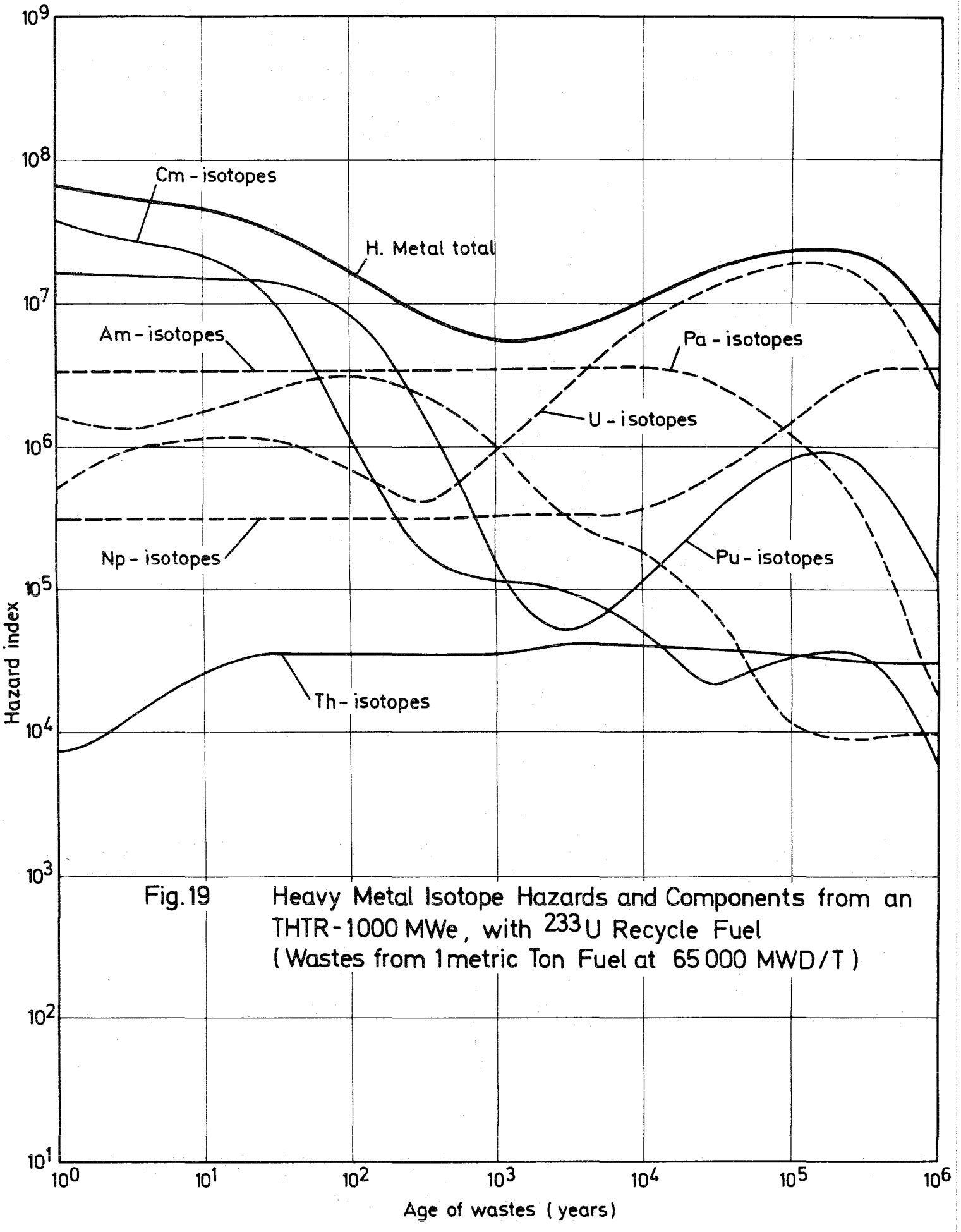


Fig. 19 Heavy Metal Isotope Hazards and Components from an THTR-1000 MWe, with ^{233}U Recycle Fuel (Wastes from 1 metric Ton Fuel at 65 000 MWD/T)

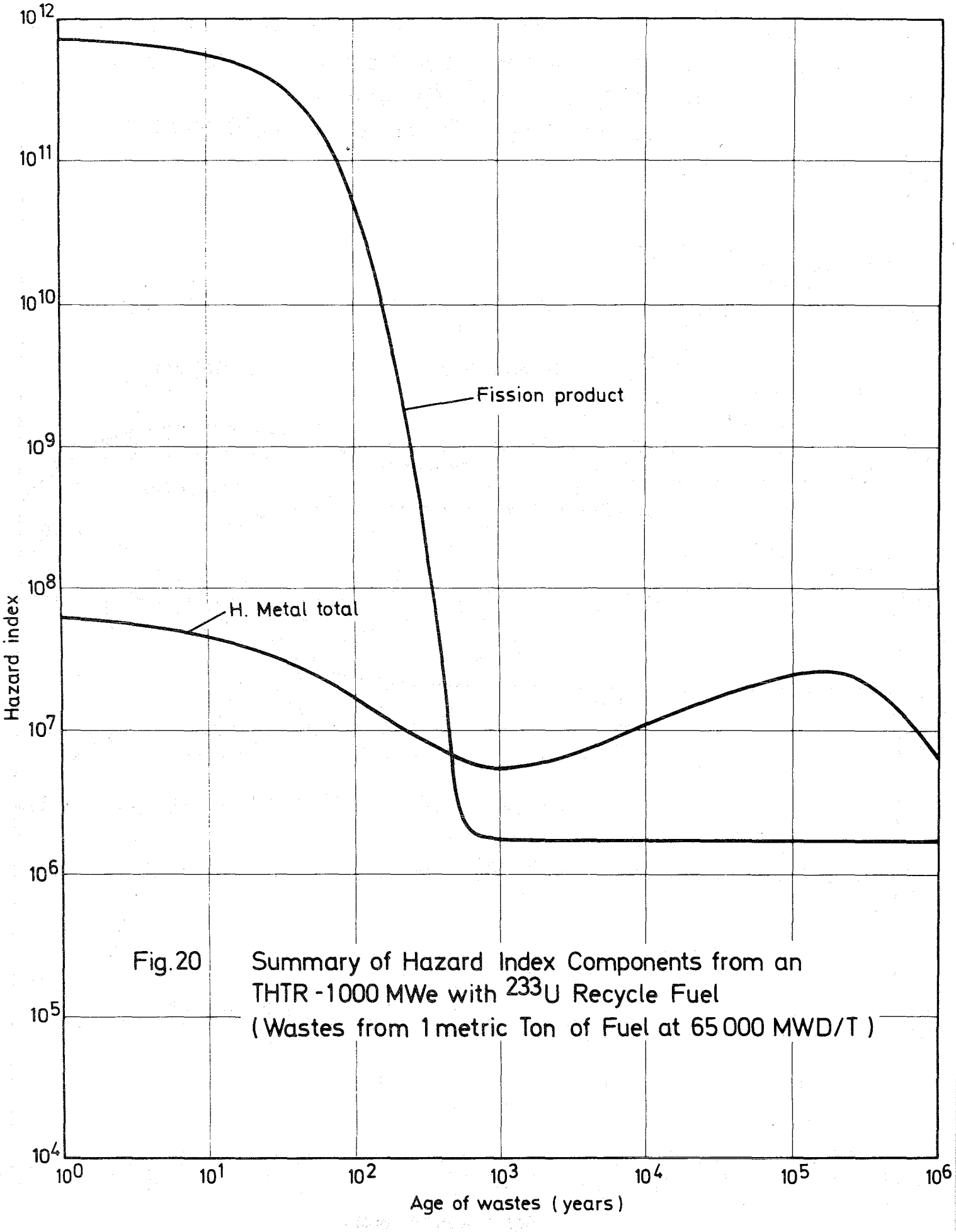
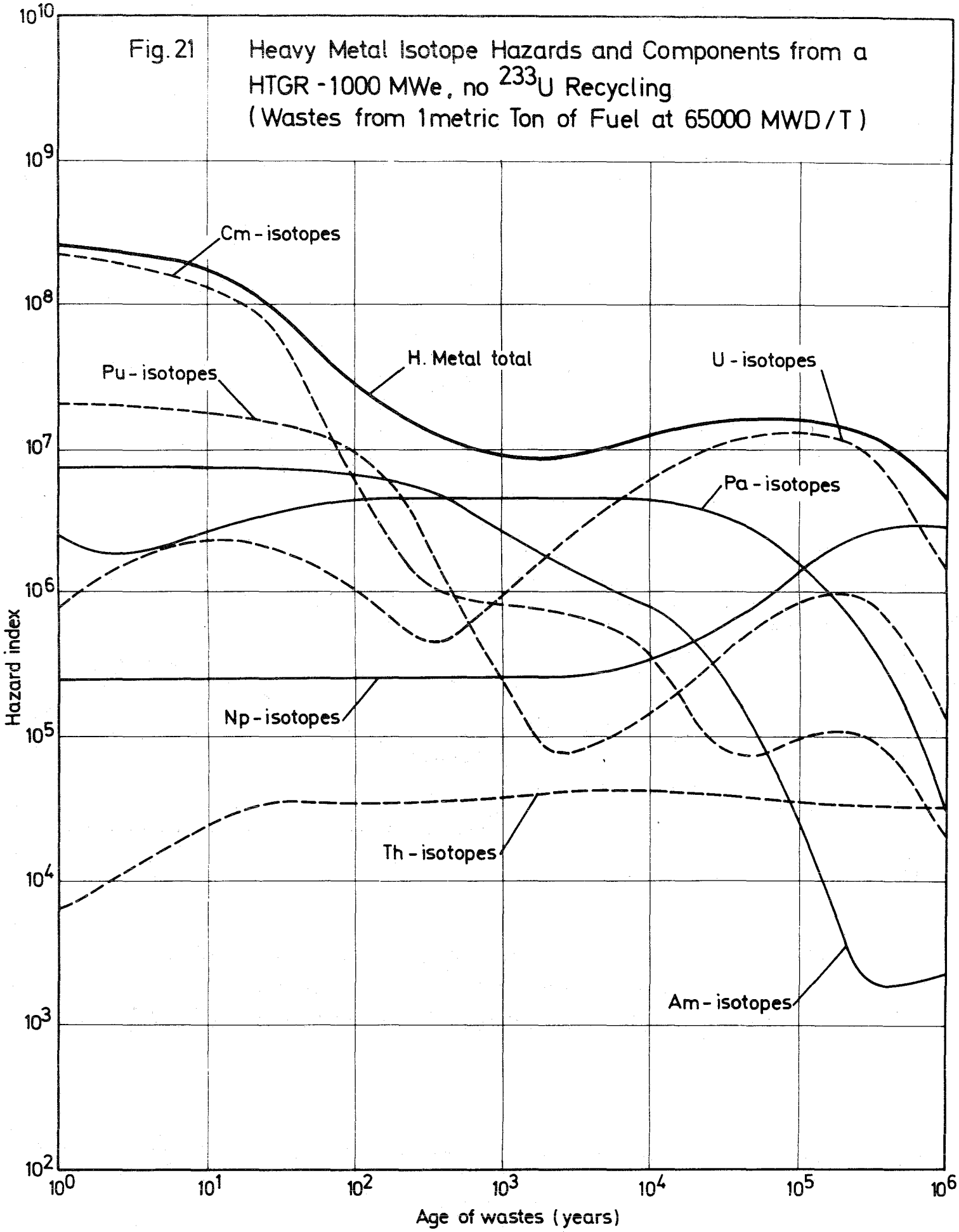


Fig.20 Summary of Hazard Index Components from an THTR-1000 MWe with ^{233}U Recycle Fuel (Wastes from 1 metric Ton of Fuel at 65 000 MWD/T)

Fig. 21 Heavy Metal Isotope Hazards and Components from a HTGR - 1000 MWe, no ^{233}U Recycling (Wastes from 1 metric Ton of Fuel at 65000 MWD/T)



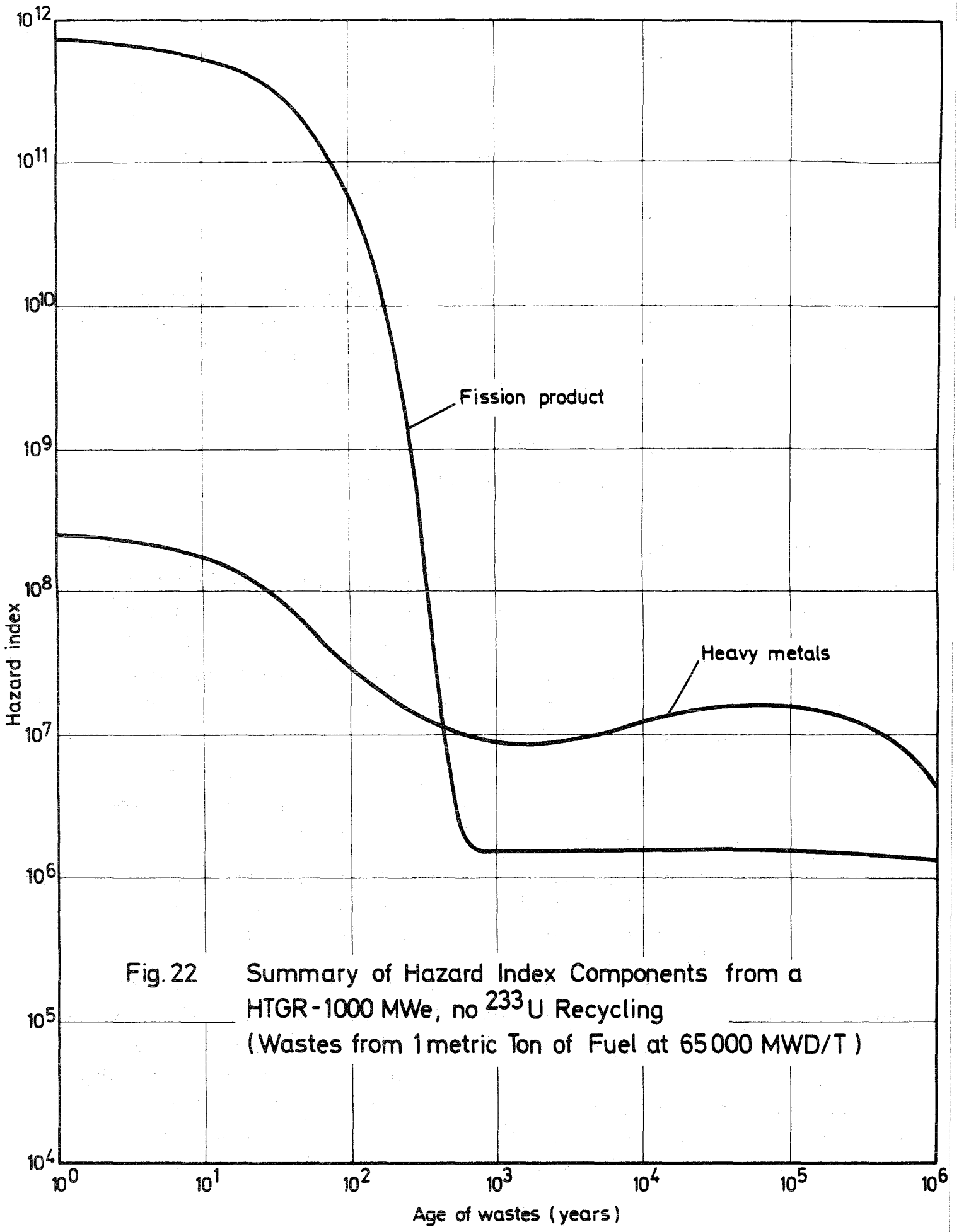


Fig. 22 Summary of Hazard Index Components from a HTGR-1000 MWe, no ^{233}U Recycling
(Wastes from 1 metric Ton of Fuel at 65000 MWD/T)

V. Waste Disposal Schemes and Associated Risks

In this section of the paper we shall discuss many of the proposed waste disposal concepts. The concepts have been organized into three categories, namely (1) ultimate disposal, (2) long-term disposal, and (3) storage. The problems associated with each scheme and the associated risks are outlined.

A. Ultimate Disposal

As explained earlier ultimate disposal of radioactive wastes refers to disposal schemes such that after disposal has been successfully accomplished it is impossible for the wastes to return to man's biosphere. The two schemes that fulfill this criterion are disposal by,

1) Nuclear Transmutation

and 2) Deep Space Disposal.

These two schemes are discussed separately below.

A1. Nuclear Transmutation

The prospect of "burning" our radioactive wastes has indeed a very attractive appeal to it. However, the feasibility and the justification of transmuting the wastes is still under heated discussion. The concept involved is that one produces a stable isotope, or one that is less hazardous, through a nuclear reaction. In considering the feasibility of the concept several criteria must be satisfied, these being:

1) The transmutation rate must be significantly faster than the natural decay rate and comparable in magnitude to the production rate of the isotope in the nuclear power reactor.

2) An overall energy balance must be fulfilled. In other words, the power derived when producing the original isotope must be greater than that required to transmute the isotope.

The proposals to date for transmuting the wastes involve the use of intense accelerator sources, fission reactors (thermal and fast) and fusion reactors. The use of intense accelerator sources, as proposed by Gregory and Steinberg /32/, appears to fall short of being able to achieve a positive energy balance as shown by Davidenko /33/. It is interesting to note that Weinberg /34/ considers the possibility of having high-powered neutron

generating accelerators in the next 25 to 50 years. The controlled thermonuclear reactor could serve as an inexpensive intense neutron source, as prosposed by Wokenhauer /35/, if and when such a reactor exists. The prospect of using the (n, 2n) reaction, which is possible with the 14 MeV neutrons from the fusion reactor, is very attractive. With this reaction one would transmute an isotope and at the same time produce two additional neutron for further transmutation reactions. However, both the accelerator and fusion reactor neutron sources can, at present, only be considered as possibilities in the distant future.

The prospect of using the fission reactor itself for transmuting its own wastes is a real possibility. It is not possible, under current reactor conditions, for the reactor to destory its own unseparated radioactive wastes since neither of the two criteria above can be satisfied. Steinberg et.al. /36/ considered the possibility of transmuting selected fission product isotopes from the radioactive waste stream, namely ^{85}Kr, ^{90}Sr and ^{137}Cs. However, since the isotopes exist in the waste only as fractions of the total element, where the weight rates of the isotopes to the element are:

	LMFBR	LWR	HTGR
$^{85}Kr/Kr$ =	~7%	~8%	~8%
$^{90}Sr/Sr$ =	~60%	~61%	~55%
$^{137}Cs/Cs$ =	~35%	~45%	~47%

it is necessary to isotopically enrich the transmutation target to achieve a minimum transmutation cost. The process would involve a recycling of the waste from the transmutation target in the fission reactor to an isotope enrichment plant when a specific fraction of the target has been transmuted. As a result of the low capture cross sections of these fission products and the flux level of 10^{13} to 10^{15} n/sec-cm^2 of present reactor designs one is not able to achieve a transmutation rate significantly greater than the natural decay rate. This applies also to other important fission product isotopes. Therefore, the fission reactor of present day design, does not appear to have the possibility for transmuting fission product isotopes.

However, the possibility of a transmutation process with fission reactors for the actinides appears to be much different for the following reasons: 1) the half-lifes of the actinides are of the order of 10^4 years and 2) their cross sections are at less an order of magnitude larger than those of cesium or

strontium. Since the transmutation process for the actinides is a fission reaction the assumption involved is that the resulting fission products are less hazardous than the actinides. In this respect the actinides, for example curium or americum, in the reactor do not differ from the uranium or the plutonium isotopes of the reactor fuel. That is, the actinides serve as a source of energy in the transmutation process and therefore do not differ fundamentally from an equivalent amount of plutonium fissions to produce the same energy.

Claiborne /37/ studied the effects of recycling the actinides through a PWR reactor. He found that the effect of the recycling did not have a significant impact on the neutron economy of commercial reactors. The reduction in the long-term hazards[1] of the waste was reduced by a factor of 40 to 200 for the case one could separate 99.5% or 99.9% of the actinides from the waste, respectively. The technology to separate the actinides (Np, Am, Cm) from the waste does not, however, presently exist. Calculations similar to Claibornes were performed for the recycling of the actinides through fast breeder reactors. The americum and curium isotopes produced in the reactor were added to the next fuel charge of the reactor. One sees from the results of these calculations, Fig.23, that an equilibrium is reached after many recycles, that is, the inventory of these isotopes becomes constant with time.

The important point here is that the value of "burning" the actinides is directly related to the degree that the isotopes can be separated from the wastes. For example, in Figs.11,15, and 16 it is seen that one gains little in a hazard reduction by removing <u>all</u> of the americum, curium and neptinium isotopes from the waste as long as the 1.0% plutonium loss remains. Therefore, the entire value of the scheme will depend on the ability to develop a technology capable of separating to a high degree (99.9999%) all the heavy metals from the fission products. From Figs.13 and 17 we see that a clean separation of actinides from the fission products will only gain us something of the order of one, or maximum two, orders of magnitude reduction in the long-term hazard due to the presences of the long half-life fission products. The most important of these long half-life fission products are, as seen in Fig. 10, in descending order of importance, ^{129}I, ^{99}Tc, ^{93m}Nb, ^{135}Cs and ^{93}Zr. Therefore to gain a significant reduction in the long-term

1) Note from Figs. 13, 17, 20 and 22 that the transmutation of the actinides would result in no significant reduction in short-term hazards.

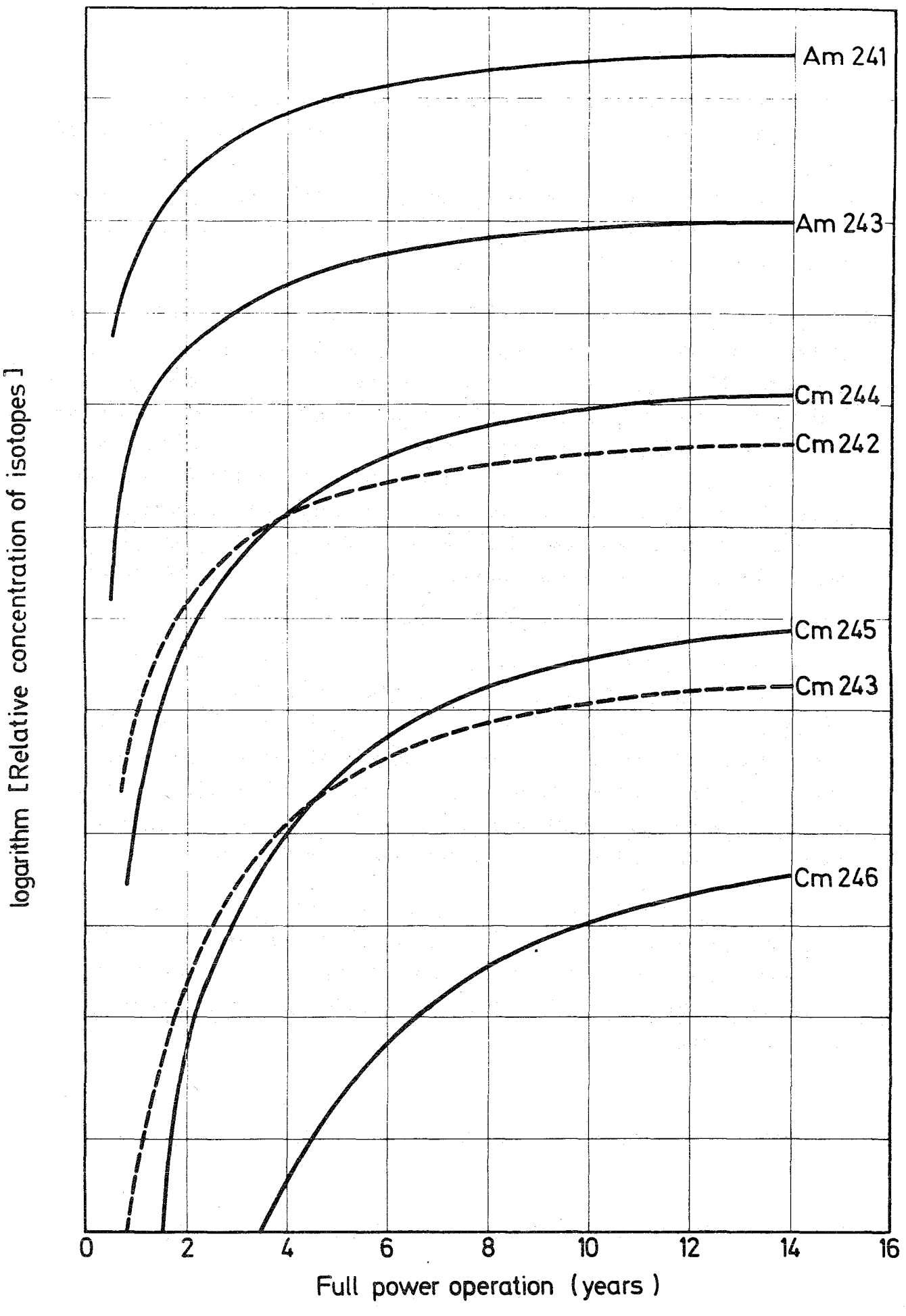


Fig. 23 Actinide Recycling in an LMFBR

hazard of the waste it will also be necessary to treat these five long half-life fission products. The essence of the argument here is that if one considers recycling the actinides one must at the same time consider a disposal procedure for the above named five fission products. The solution to the problem of long-term hazard reduction by transmutation hinges on a solution to both problems.

Finally,it must be pointed out that the subsequent separation of selected isotopes from the waste stream is itself not without associated problems even if we can cleanly burn the selected isotopes. As evident from fuel reprocessing one would generate large volumes of contaminated chemicals, water, and air which must be properly treated. Therefore,the risks associated with the extra handling and process releases involved in separation of the isotopes from the waste stream and subsequent transmutation must be weighed against the reduction in a future risk, due to the presence of these isotopes,by use of the concept of social rate of discount explained in section III. C.

To get an impression of the size and importance of the problem the production of the important actinide isotopes, other than uranium and plutonium, in representative reactors is given in Table 11 . In combination with the nuclear energy strategy given in section II of this paper the annual and accumulated production of neptunium, americium, and curium in the BRD are given in Table 12 . The annual production of these isotopes is shown graphically in Fig. 24 .

For the purpose of illustration let us assume a concept for the transmutation of these isotopes. We could reasonably assume that the isotopes would be concentrated in a special element, or elements, to be placed in the reactors from which they originated. The system might look like the following.

Table 11 Actinide Production in Representative Reactors
(grams per ton of fuel reprocessed at 150 days)

Isotope	LWR[1)]	LWR[2)]	LMFBR[3)]	LMFBR[4)]	THTR[5)]	HTGR[6)]
Neptunium	500	368	263	271	1260	1090
Americium	153	774	959	259	11	36
Curium	43	522	22	5	3	19
Protactinium	--	--	--	--	52	68

1) PWR operating with equilibrium uranium fuel cycle to 34,000 MWD/T burnup

2) PWR operating with Pu-recycle fuel (19% of fissile charge is plutonium) to 34,000 MWD/T burnup

3) Sodium cooled fast breeder reactor with core and blanket mixed proportionally at reprocessing, light water reactor plutonium fuel, average burnup 34,000 MWD/T

4) Sodium cooled fast breeder reactor with core and blanket mixed proportionally at reprocessing, fast breeder reactor equilibrium plutonium fuel, average burnup 34,000 MWD/T

5) high temperature gas cooled thermal reactor operating with equilibrium ^{233}U recycle fuel to 65,000 MWD/T burnup

6) high temperature gas cooled thermal reactor operating with equilibrium high enriched (93%) ^{235}U fuel cycle to 65,000 MWD/T burnup

Table 12 Production of Neptunium, Americium, and Curium in the BRD

	1980	1990	2000	2010
INSTALLED CAPACITY, GWe	20	63	155	316
Annual Production, kg/year				
Neptunium	178	627	1579	3158
Americium	55	229	1018	2789
Curium	15	54	135	270
Accumulated, kg				
Neptunium	697	4485	15221	38383
Americium	213	1465	7193	25412
Curium[1)]	60	386	1307	3291

1) includes daughter products (plutonium)

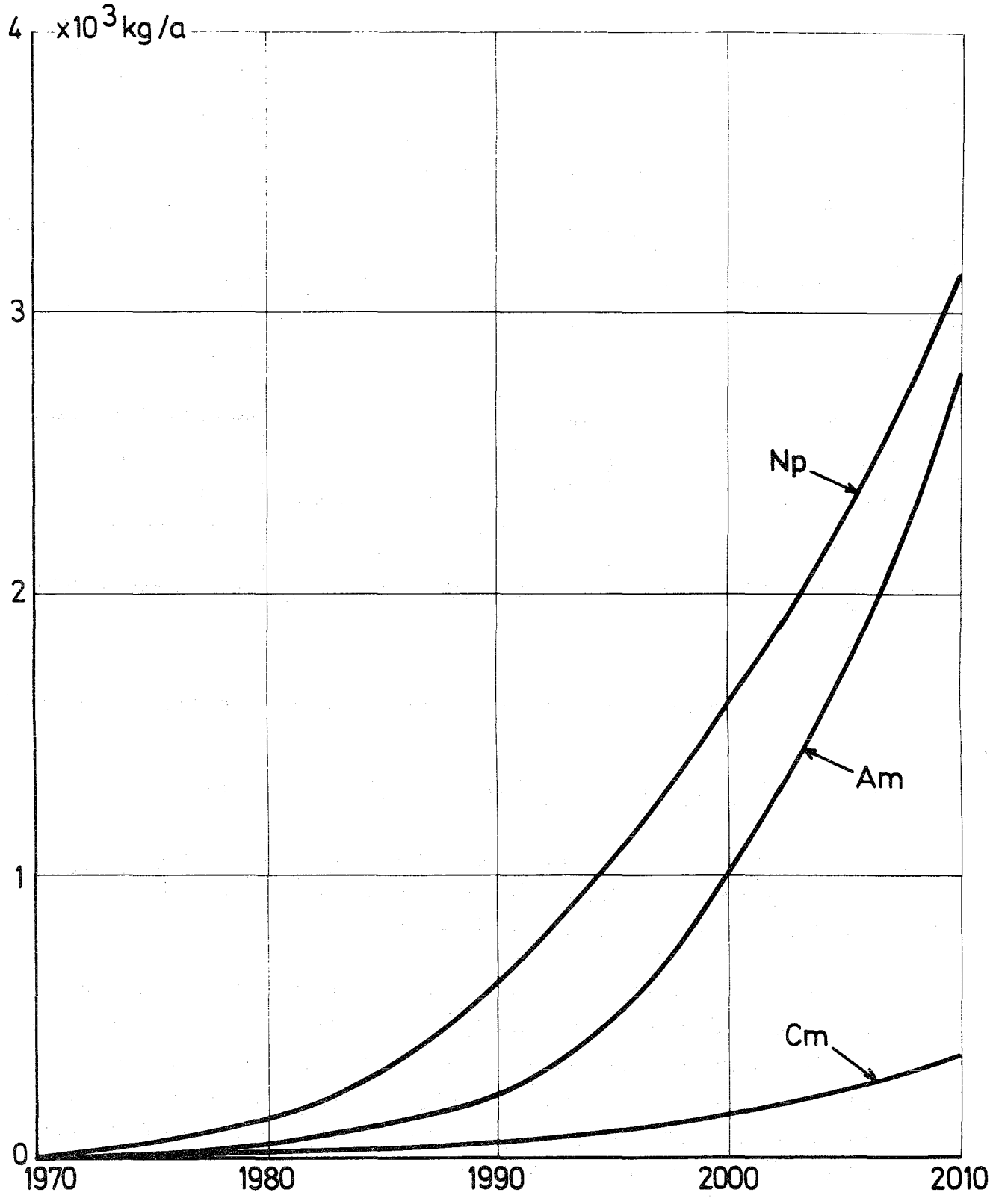
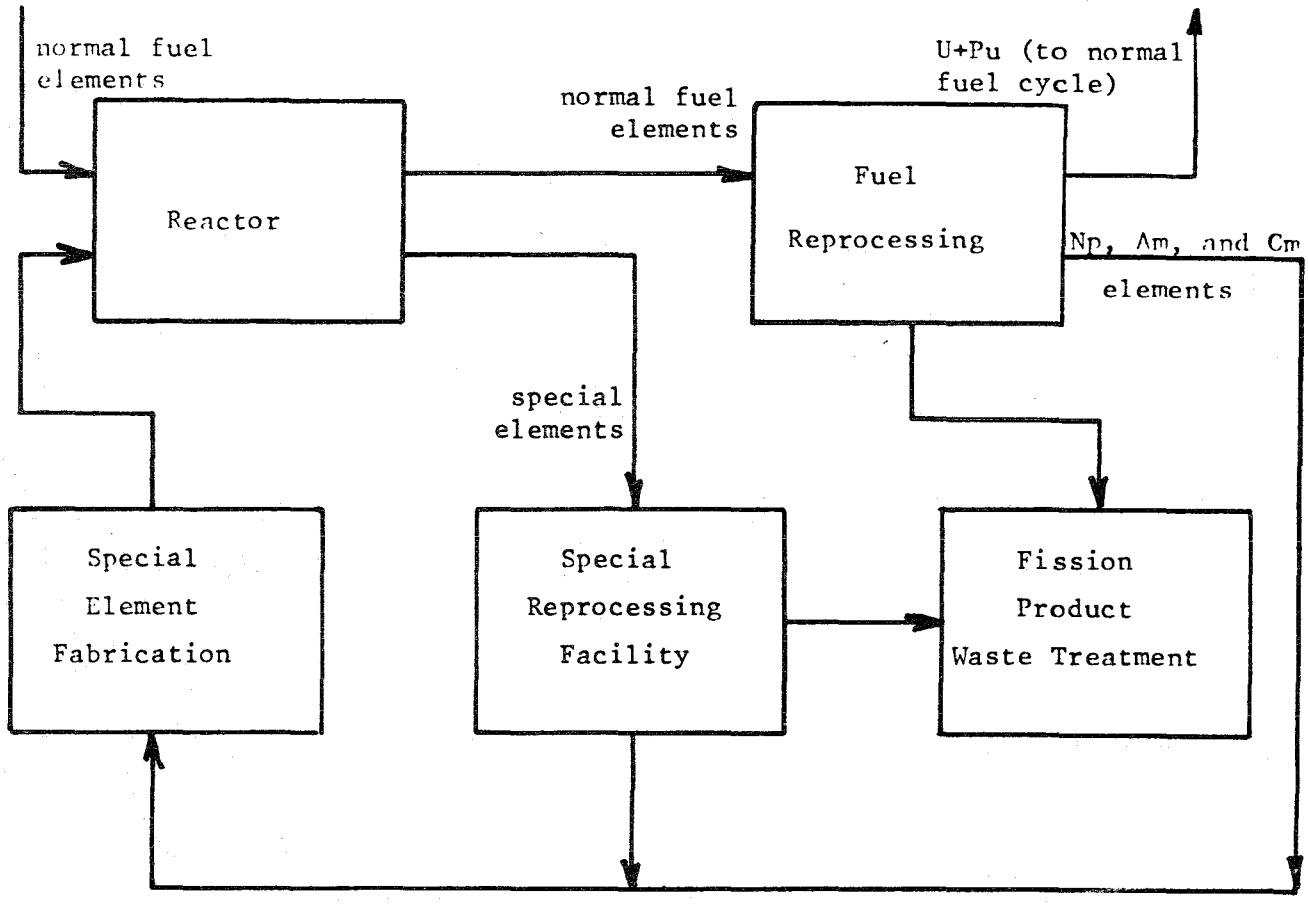


Fig. 24 Annual Production of Np, Am and Cm in the BRD



Possible Nuclear Transmutation System for Actinides

In this scheme the left-hand loop of the special cycle could be broken away from the reactor and fed through any other type of transmuting device, e.g. an accelerator.

Due to the high neutron source associated with the curium isotopes, it is reasonable to assume that special facilities could be needed to handle the special elements containing the actinides to be transmuted. In addition, it would probably not be desirable to mix the actinides with the fuel mixture of the reactor and suffer an overall fuel handling and fabrication penalty.

In the above scheme we assume optimistically that the actinides can be separated from the fission products with a decontamination factor (DF) of 10^4 for all the isotopes and that the waste stream in the reprocessing facility can be decontaminated to a factor of 10^7. In addition, we assume that the special elements can remain in the reactor

or other transmuting device to the point that 10% of the atoms present have fissioned. For metallurgical reasons the special elements would have to be removed after 10% burnup and reprocessed. Therefore, ten passes through the reactor would be needed to fission all the atoms originally present in the element in its first loading into the reactor.

Therefore, with DF = 10^4 and ten passes around the cycle the cumulative loss to the fission product waste stream would be

$$10^4 \times (10/10 + 9/10 + 8/10 + \ldots . 1/10) = 0.00055$$

of the actinides originally associated with the fission products. The composition of the actinides would be tilted toward the higher end of the isotope chart due to the neutron capture reactions. This, however, would probably not be an important point except possibly in the handling procedures due to the higher spontaneous neutron fission source. Notice that if the DF was not large, and the number of passes around the cycle was not kept to a minimum, the cumulative loss to the waste stream would be large due to the additive effect.

With these data and the accumulated production of neptunium, americium and curium in the BRD to the year 2010, there would be approximately the following quantities in the fission product wastes and the biosphere.

Element	In F.P. Waste (kg)	In Biosphere (gm)
Np	21.1	2.1
Am	13.9	1.4
Cm	1.8	0.2

The DF = 10^4 for the actinides in the wastes would put the heavy metal hazards, see Figs. 10 and 11, at 10^3 years between that of ^{129}I and ^{99}Tc for the PWR's and the LMFBR's. The THTR and HTGR heavy metal curves lay flatter and lower than those for the PWR's and the LMFBR's, therefore in these cases the gain is slightly larger but qualitatively not different.

Nevertheless, assuming that the same decontanimation factors are also obtainable in the normal fuel reprocessing facility, regardless of

whether we aspire to transmute the actinides or not, the decision to transmute the actinides would result in roughly 5.5 times the amount of heavy metal isotopes deposited in the environment had we not chosen to transmute the actinides. The question is, "Is this equal to, or less than, the achievable reduction of future risks as viewed from the present?".

A2. Deep Space Disposal

Since the days of dwindling interest and funding for space research and exploration, the use of space technology to solve the problem of radioactive waste management has been increasingly advocated. Studies evaluating the alternatives of extraterrestrial transport of the wastes have been published /38, 39, 40/. The two overriding problems with this type of a disposal concept are the cost and short-term safety (safety in the sense of todays hazards in contrast to hazards to future generations). Cost is directly a function of the launch trajectory (earth escape or solar impact) and the waste-to-payload ratio.

To put the problem in perspective it is necessary to evaluate the amount of "rocket power" necessary to dispose of the wastes that will be produced in the BRD. Up to the year 2010 it is estimated that approximately 88,000 tons of fuel will have been reprocessed. (See Table 4 in Section II. C3.) This fuel will have an average burnup of about 34,000 MWD/T. The composition of the waste will include about 36kg of fission products and about 10 kg of heavy metals per ton, assuming a 1.0% loss of uranium and plutonium during fuel reprocessing. This means that up to the year 2010, 4050 tons of waste will have been produced. In the year 2010, using the same figures, the annual production rate of waste will be 415 tons. The payload of a Saturn V rocket for solar orbit is about 35 tons. Assuming optimistically a waste-to-payload ration of 25%, one would need four Saturn V launches a month by the year 2010 to handle the total wastes of the BRD. It would be, of course, absurd to consider disposing of all the wastes in space. A significant fraction of the waste is either stable or of fairly short half-life ($t_{1/2}$ < 10 years). If we partition the fission product wastes into elements having radioactive isotopes with half-lifes greater than 10 years the total amount, per ton of fuel at 34,000 MWD/T, would be only about 14kg for both the PWR's and the LMFBR's. This in essence would halve the number of launches needed. Nevertheless, it still is a significant number, assuming of course that the waste-to-payload ratio could be achieved.

Preliminary results of Platt and Ramsey /39/ indicate that one would suffer a prohibitive penalty for materials requiring substantial shielding for gamma emissions or Bremstrahlung. It appears, therefore, that payloads would have to be restricted to partitioned fractions of the radioactive waste consisting of the transuranics contaminated by 0.1 to 1.0% of the fission products. Based on this result the ultimate disposal schemes appear to be suitable, with present day technology, only for the "ultimate" disposal of the actinides. Using the data of Table 12 in section V. A1, the annual production of neptunium, americium, and curium by the year 2010 is roughly 6200 kg. In addition, 1% of the fission product wastes would amount to about 3250 kg more. Assuming again an optimistic waste-to-payload ratio of 25% one would need slightly more than one launch per year for the BRD in the year 2010 to dispose of the annual production of actinides by placing them in a solar orbit.

The risk from waste disposal by extraterrestrial transportation can be assigned to five separate segments of the launch procedure. These are

1) prelaunch handling accident,
2) launch-pad mishap (explosion of rocket),
3) mishap during ascent of the rocket between launch-pad and the approach to orbit,
4) failure to orbit,
and 5) reentry into the atmosphere after successful orbit has been achieved.

A more detailed discussion of the safety problems associated with the launching into space of nuclear materials is given by Branch and Connor /41/. They consider hazards control for satellite nuclear auxiliary power (SNAP) devices. These are basically isotope and small reactor units, total radioactivity of which would be of the order of magnitude of a megacurie (MCi) or less. They suggest that these power units be designed to burn up upon reentry since they feel this would result in the lowest possible hazards. In fact,on on April 24, 1964, a Transit navigational satellite with a SNAP unit

(designated SNAP-9A) containing 17 kCi, or about 1kg of ^{238}Pu, failed to achieve orbit /42/. Subsequent measurements of ^{238}Pu in the atmosphere confirmed that the SNAP-9A unit had burned up upon reentry into the atmosphere. This accidental release of ^{238}Pu almost tripled the global deposit of this plutonium isotope by 1970. The global deposit of ^{238}Pu from weapons testing is estimated to be about 7.7 + 0.9 kCi.

However, due to the larger radioactivity of waste disposal launches, it would not be desirable to design for burn up upon reentry. Therefore the design of the containers to survive reentry and impact is considerably more difficult. In addition, it would be desirable for the containers to withstand melting as long as possible after impact to facilitate after accident clean up operations. An analysis of these types of waste containers is given by Van Bibber and Paker /43/.

As in the preceding section, the assumed additional hazards (short-term) of disposal by extraterrestrial transportation must more than compensate the reduction in the long-term hazards of the wastes, as evaluated by a social rate of discount concept if the concept is to be used. Assuming that the actinides could be removed from the waste stream with a decontamination factor of 10^4, as in the preceding section, than the long-term hazards of the heavy metals would lay between that due to ^{129}I and ^{99}Tc (see, for example, Figs. 10 and 11).

B. Long-Term Disposal

To reiterate the meaning attached to the term long-term disposal we refer to disposal on or in the earth in such a manner that the wastes are, for all practical purposes, nonretrievable. The risks associated with such schemes is that it is conceivable, although with a small probability, of the wastes returning in an uncontrollable manner to man's biosphere.

The schemes that are considered here are the following:

1) Disposal in salt deposits
2) Disposal in a nuclear cavity (Plowshare Concept)
3) Disposal under an ice sheet (Antarctica)
4) Disposal in the earth's crust by self-burial.

This list is by no means exhaustive. For example,there are other proposals such as disposal in tectonic sinks, in vaults excavated in crystalline rock, and by hydraulic-fracturing. However, while these and others purposed schemes differ in detail from the four listed above,the problems encountered are not significantly different than the ones to be resolved in any of the four schemes above.

B1. Disposal in salt deposits

Disposal of high-level waste in salt deposits has been studied intensively both in the United States and the BRD. In the United States the concept is to use bedded salt deposits /44/ and in the BRD salt domes /45/. In either case the basic design is similar.

The advantages of salt over other rock types as a medium for the disposal of waste are listed here in the following:

(1) Isolation from water over a long period
(2) Plastic properties under hydrostatic pressure
(3) Good thermal properties
(4) Ease of mining
(5) Generally in zones of low seismicity.

The primary disadvantage of salt is its high solubility in water. However the very presence of salt in massive bodies beneath the ground attests to the fact that salt has, in general, been isolated from circulating ground water. For any form of geologic storage, it is probably through transport in ground water that buried wastes would most likely come into contact with man's environment.

The procedures involved in the disposal of wastes in salt deposits are the following:

(1) Interim storage as a liquid
(2) Waste converted to an encapsulated solid
(3) Interim storage as a solid
(4) Transported to site of a previously excavated salt deposit
(5) Waste lowered through a shaft into the mine
(6) Transported in a carrier to a room in the mine and placed in preexcavated vertical hole in the floor.
(7) Crushed salt is put in the hole to fill it for shielding (~2m).
(8) The process is continued until all the holes in the room are filled. The room will then be backfilled with crushed salt.

The possible arrangement of the waste cylinders in such a concept is shown in Fig.25. The distance between the waste cylinders in dictated by the maximum allowable salt-temperature between the cylinders. The heat generation rate of the cylinder is determined by the volume fraction of waste to inert solid composing the cylinder, and the age of the waste (see Tables 5 to 10).

The solification of high-level waste, a step inherent in the concept of disposal in salt, is under intensive investigation around the world today. The procedure under investigation in the BRD has been recently reviewed by Krause /46/. It must be mentioned that the solidification processes generally generate low-level aqueous waste liquid which must be treated in a separate system and some of the radioactive elements are released with an effluent gas stream.

Retrievability of the waste cylinders from the salt is in principle, possible, but there is no doubt that it would be an extremely difficult operation. Conceivable waste containers could not be expected to remain intact more than a few years. If it were necessary to retrieve the waste cylinders, they would have to be mined out. For all practical purposes the waste should be considered as nonretrievable for our calculations of potential hazards.

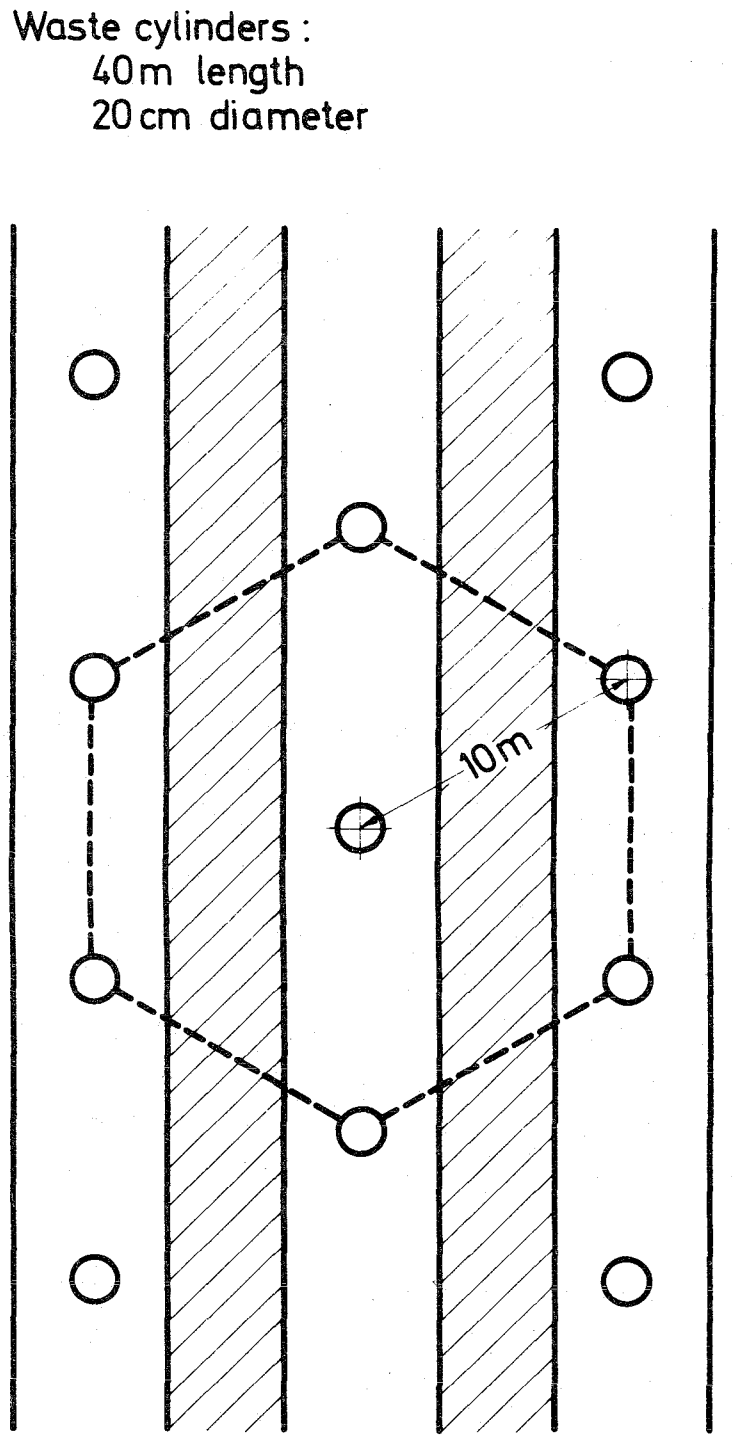


Fig. 25 Arrangement of Waste Cylinders in Salt Deposit

Definition of "Zero Point" in Time

For the consideration of the safety, or risk to the public, of this disposal concept one is faced with a very difficult problem; namely that it is necessary to consider a very long time span, equal to that of a geological time scale. A very important question at this point is: How long a time should we consider the waste as dangerous? Obviously not to the point that the last atom has decayed. Therefore it may be convenient to define a point, between the time the wastes are buried in the salt and infinity, after which time we would not consider the waste as dangerous. A possible means of doing so may be by comparing the wastes to existing natural radioactive sources, as uranium and thorium ore deposits, as originally purposed by Bell and Dillon /47/.

To make such a comparison we use the solidification process as reviewed by Krause /46/ and the spacing scheme shown in Fig.25. Directly associated with the wastes of one ton of fuel (see Tables 5 to 10) is 220 m^3 or 505 tons of salt.

This value is calculated by assuming 80 liters of glass for the waste of one ton of spent fuel at a burnup of 34 000 MWD/T. From simple geometry and the data in Fig. 25 one can calculate the associated quantity of salt radially outward from the waste cylinder. Using the definition of the hazard index as given by Eq. (1) in Section IV one finds /47/:

(1) 1.0 gm of natural uranium in equilibrium with its daughters

$$HI = 15.1\ m^3 H_2O/gm\ U$$

(2) 1.0 gm of thorium in equilibrium with its daughters

$$HI = 3.78\ m^3 H_2O/gm\ Th$$

Assuming an average concentration of uranium and thorium in the earths crust of 4 ppm and 12 ppm, respectively, one computes a hazard index for 505 tons of average earth's crust of

$$HI\ (ave.\ earth's\ crust) = 5.35 \times 10^4.$$

A typical uranium ore deposit is generally of the order of 0.2% U_3O_8. Thorium is closely associated mineralogically with uranium and is generally found in uranium ore. We shall assume a Th/U ratio of one. Therefore, 505 tons of typical uranium ore would have a hazard index of

$$HI\ (uranium\ ore) = 1.68 \times 10^7.$$

Uranium and thorium are present in a higher concentration in mineral monazite which occurs in beach sand in some areas of India, Brazil, Malaysia, and the southeastern United States. India monazite has a ThO_2 content of 8 to 10.5% and uranium content of 0.3 to 0.4%. Typically 2 to 2.5% of the sand is monazite. Using the lower value of these figures a hazard index for the sand is calculated to be

$$HI(\text{monazite}) = 3.04 \times 10^6.$$

Another possible consideration in this respect would be to compare the amount of water necessary to dissolve the 505 tons of salt to a potable concentration of 500 ppm NaCl. This would be 1.01×10^6 $m^3 H_2O_3$ or a value between the HI (ave. earth's crust) and HI(uranium ore).

From Fig. 10 and 22 one sees that the point equal to the HI(uranium ore) is reached at about 10^4 years for all the waste types. One would say that roughly 10^4 years is a time span over which the wastes could be considered dangerous.

For comparative purposes we may also use what was believed to be a natural fission reactor in the uranium ore deposits of Oklo in Africa, as reported in Atomwirtschaft /48/ and Energie Nucleaire /49/. The low content of ^{235}U ($\sim$ 0.6%) in the uranium, which is significantly below that which is generally found, led scientists to an investigation. The conclusion of the investigation was that a chain reaction, as in nuclear reactors of today, had taken place. From isotope correlations it was established that the original enrichment of the ore was about 3% ^{235}U about 1.74×10^9 years ago. With a high concentration of uranium in the ore and the possible presence of water, a chain reaction over a long time period (> 10^4 years) took place. The chain reaction is believed to have discontinued more than 10^8 years ago. Present day concentration of the ore, at the point of greatest ^{235}U depletion, is 14.9 gm U/100 gm minerals.

A hazard index for this "reactor" and its wastes was calculated, in the same manner as explained above, from the time the "reactor" was operating until today. The total hazard index in the last 10^8 years has decreased slowly, dropping only slightly more than an order of magnitude. For 505 tons of ore today the hazard index is

$$HI(\text{Oklo}) = 1.14 \times 10^9.$$

From the curves for the wastes from the various reactor types one see that this hazard index is passed in all cases in less than 300 years.

The above arguments assume that the most probable accident for a disposal scheme involving waste burial in salt deposits is an entrance of water to the burial cavern. As is discussed in the following this is considered to be the most probable accident and has been so designated in other works /50/. Also assumed is that once the water has entered the burial cavern it will flow out again. This assumption is not with solid foundation since, as proven in salt mining experience, water in a flooded salt mine generally remains in the mine. In addition the solidified wastes are not readily soluble. Leach rates for glass, which of course depend on temperature, chemical composition of the glass, element leached, and age of the glass, have been measured to be of the order of magnitude of 10^{-4} to 10^{-7} $gm/cm^2/day$ /51/. Similar arguments can also be considered for the uranium ore deposits and their calculated hazard index. For example, monazite sand is among the least soluble of naturally occurring materials. Very roughly one could say these effects compensate each other so that our comparison above is not entirely false.

Accident Situations in the Waste Facility

The initiation of an accident in a waste facility could result either from intentional, or unintentional, human activities or from geological processes. In this context it is convenient to consider accident situations in two different time periods. First is the geologically short time period when the waste facility is in operation (waste being brought into the facility) and the geologically long time period when the disposal facility is sealed. The factors involved in these two time periods are not identical.

1) During Operation of Waste Facility

To logically relate the factors involved in the release of radioactive material from the waste facility a fault tree, shown in Fig. 26, was constructed. One notices that there are two basic types of release initiating accidents, the more or less random accidents due to human error and the geological accident (flooding of the mine) which, while initiated due to human error, would not be considered as random. Except for the flooding of the mine the other accidents could be easily controlled and contained through proper instrumentation and immediate personnel actions. The flooding of the mine can be effectively engineered against, and would have to be considered as a highly improbable accident. However, should such an accident occur it would be difficult to bring the facility back into operation and/or guarantee

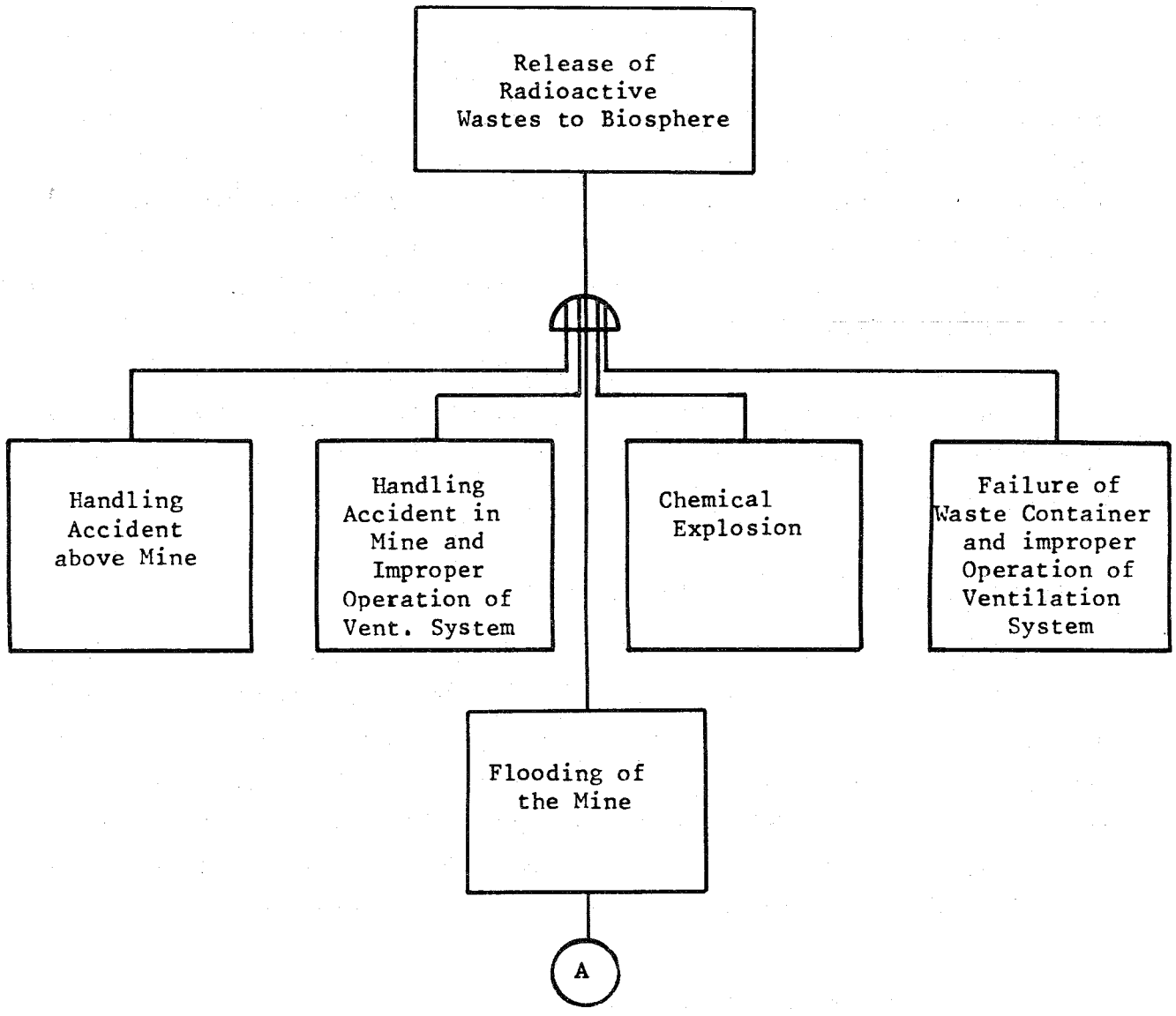
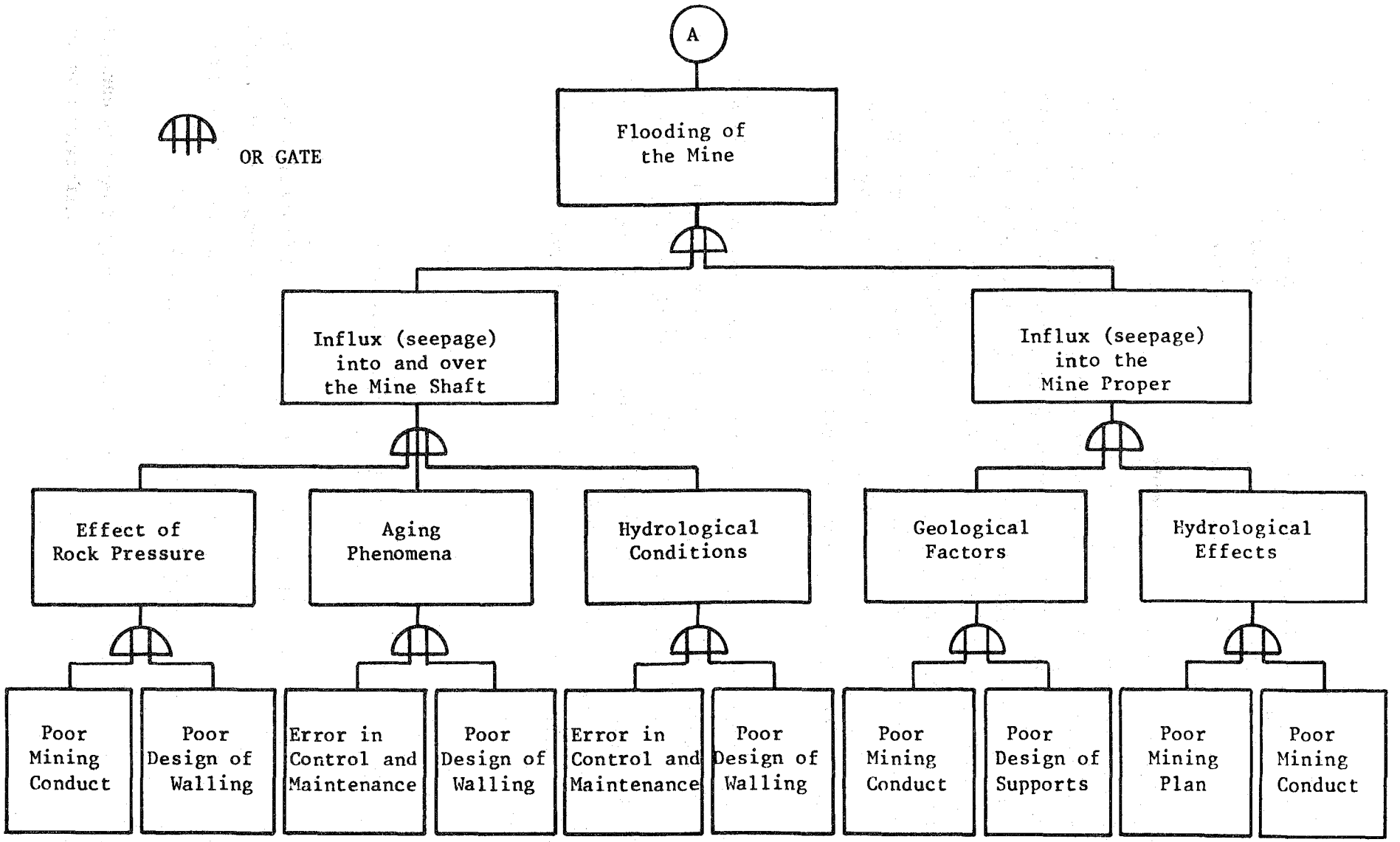


Fig. 26 Fault Tree of Salt Mine used for Storage of Radioactive Wastes (during operation of the mine; before closure and sealing)



(walling: supports of sides and ceiling in mines)

Fig. 26 con't Fault Tree of Salt Mine used for Storage of Radioactive Wastes
(during operation of the mine; before closure and sealing)

the safety of the wastes already disposed. If the mine were flooded it is not likely that the water would flow out again. It could be pumped out and treated as contaminated water. However, if the water were to come into a "hot" area where wastes had already been deposited, waste cylinders could be exposed if a chamber were to collapse.

To attach probabilities to the events in the fault tree of Fig. 26 is a very difficult job since to date there does not exist much experience in the disposal of high-level radioactive wastes in salt deposits. The use of statistical data for the flooding of salt mines would be unfair since the precautions to prevent the flooding of a salt mine prepared for the disposal of radioactive wastes would be considerably more extensive than those for a normal salt mine. It is probably reasonable to assume here that the waste disposal facility would be so designed that if flooding did occur it would not be possible for the water to flow to the region containing exposed chambers in active use for disposal.

The handling of waste cylinders at the disposal facility would, in some ways, be comparable to the handling of spent fuel elements at nuclear power plants or fuel reprocessing facilities. The waste cylinders and spent fuel elements both must be shielded against their intense radioactivity and cooled because of their inherent large heat source. For example, the class 6 and 7 accidents of the USAEC for nuclear power plants are the following:

1) Class 6 - Refueling Accident Inside Reactor Containment

2) Class 7 - Accidents to Spent Fuel Outside of Reactor Containment.

If we substitute Waste Disposal Facility for Reactor Containment and waste cylinder for fuel element in the above we have a very comparable situation. The same applies for fuel reprocessing facilities where the class 4 accident is "Fuel Handling Accidents Involving Cladding Failure".

From experience gained in the operation of nuclear power plants we can get an idea of the frequency of human errors which lead to significant events.

A compilation of data collected for the three years 1970-1972, as published in Ref. /52/, gives an average probability of an event per facility-month of 0.05 due to human error. As defined here a significant event due to human failure includes the failure to follow procedures, which in most cases would not endanger the public in any manner. A waste disposal facility would certainly not approach the complexity of a power reactor system. Therefore, we shall use a factor of 10 to describe the difference in complexity between the two types of facilities. In addition, we shall assume that only one in a hundred human errors would result in a subsequent release of radioactive material. Therefore, the probability of a release event occurring due to human error is 0.00005/month, or 0.0006/year. It has been observed that on the order of half of the accidents occurring in nuclear reactor power plants can be traced to human error. Therefore to adjust our probability of an event occurring to include all causes we multiply by 2, or P_r = 0.0012/year.

Similar accident probabilities have also been estimated, based in part on past experience, for fuel reprocessing facilities. The probability estimated in a fuel reprocessing facility /53/ for fuel handling accidents involving cladding failure is P_r = 0.05 to 0.075/year. In a 1000 MWe reactor approximately 30 tons of fuel are handled per year, where as in a large reprocessing facility on the order of 1500 tons of fuel are handled per year, or 50 times as much. Basically this can be interpreted as the accident is about 50 times more likely to happen in a year at the fuel reprocessing facility than at the power plant. Adjusting our estimated probability from the power plant for this factor,

$$P_r = 0.0012 \times 50 = 0.06/\text{year}$$

we find that it fits relatively well with the probability estimated for the fuel reprocessing facility.

After the spent fuel elements have been reprocessed and the wastes treated for disposal the reduction in volume/weight would be approximately 20% of that associated with the originally fuel elements. If we assume that a waste disposal facility would handle the wastes from five reprocessing facilities (7500 tons of fuel/year) the amount of handling, and probability of a handling accident in a year, would be our estimated value, P_r, above.

2) After closure of Waste Facility

After the waste facility has reached its designed capacity it will be permanently sealed and, for many years thereafter, guarded against unintentional entry. The ground surface of the facility will probably also be monitored for many years after closure for the purpose of detecting releases. However, one could not expect these activities to continue for the time period of 10,000 years, defined in a previous section as the period of danger for the wastes. During such a long period of time geologic processes can play a very important role along with other factors. An excellent collection of data on geologic processes relevant to the disposal of waste in geologic formations is given by F. Gera and D.G. Jacobs /54/.

A consideration of the events which could lead to the release of radioactive materials from the waste disposal facility over very long periods of time has resulted in the fault tree shown in Fig. 27. This fault tree was not intended to serve for the calculation of the probability of the top event. This is presently not possible as geology has been a science with very limited predictive capabilities. However, from the fault tree one is able to get a logical ordering of the events.

Many of the factors in the fault tree can be designed against by careful selection of the location of the waste facility. In addition, the geologic processes are very regional dependent so that one cannot, in general, make a statement concerning the possible effects of these processes. Under the assumption that what has not happened in the past will not happen in the near future ($\sim$ 100,000 years), a site would be selected that is tectonic stable, experiences a low rate of erosion, and has no record of volcanic activity. This assumption is not necessarily valid in the world of today in which the influence of man on his environment is ever increasing. A case in point is an incident that occurred several years ago in the mountain region near Denver, Colorado, USA. The injection of liquid chemical wastes into underground caverns was halted when earth tremors were experienced in an area that had had no previous records of such tremors.

It is, nevertheless, instructive to discuss the range of the variables in the fault tree.

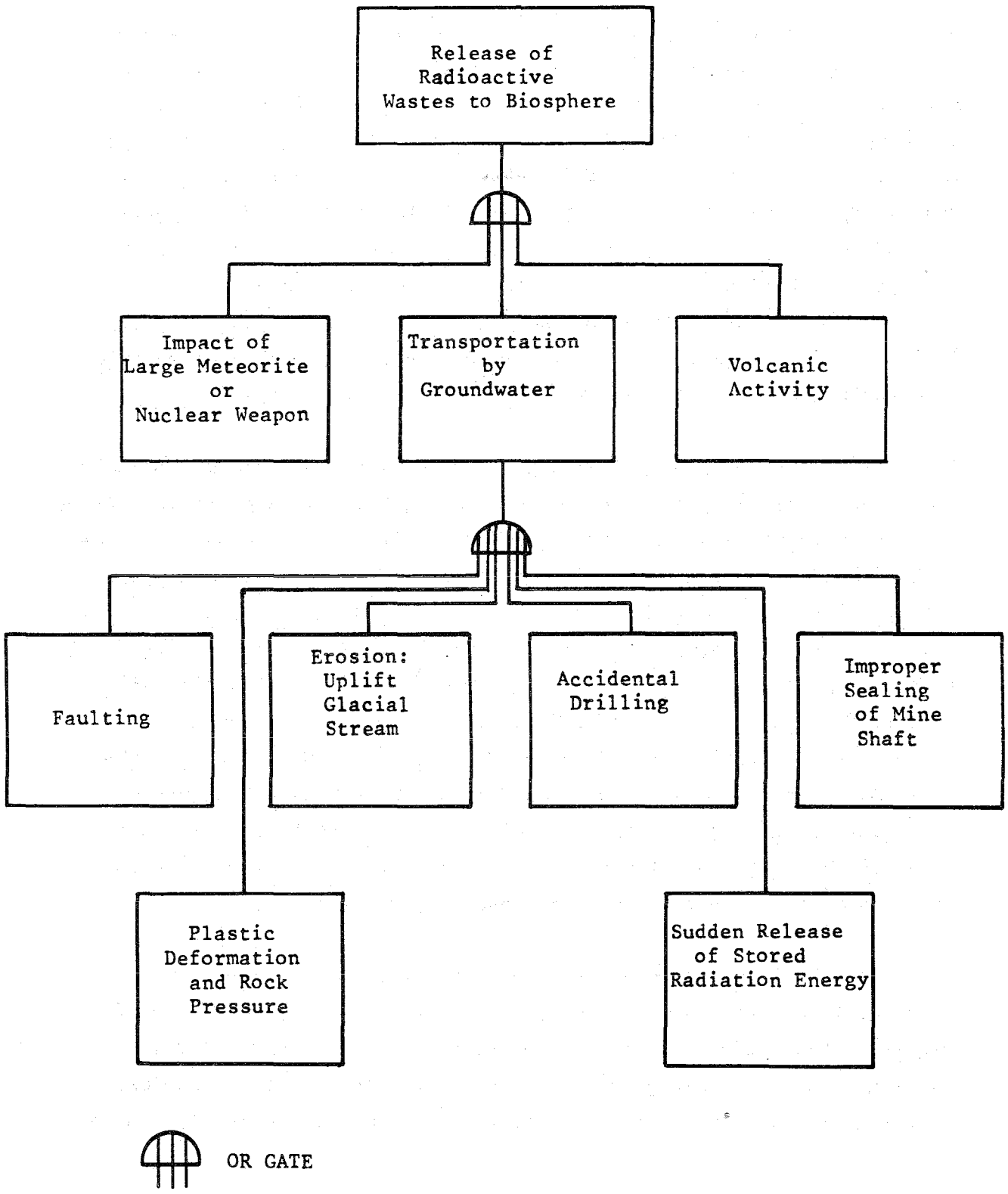


Fig. 27 Fault Tree of Salt Mine Used for Storage of Radioactive Wastes (after closure of the mine)

Erosion is controlled by climate, relief and the lithological nature of the materials subject to erosion, and also by human activity. Erosion rates vary from 10-100 cm/1000 years with most of the data in the range 20 to 50 cm/1000 years. It is believed that glacial erosion has cut to depths of 600 to 1300 m. For example, the present rate of erosion of the Muir Glacier in southern Alaska is 2000 cm/1000 years. Glaciers are known to move at an average rate of about 10 m/year. The last glacial retreat occurred only about 12,500 to 15,000 years ago. The next ice ago is expected to occur in about 10,000 to 50,000 years from now. Uplift seems to be very time dependent, but the average over long periods of time (10,000 years) runs about 3 to 10 mm/year.

The impact of a large meteorite could be considered as a random process and, therefore, the probability of the incident could be estimated. Blake /55/ has calculated the probability of meteorite impact on the earth as a function of meteor weight. If we consider only those meteors having the capability of cratering at least to 200 to 300 m (meteors $\geq 10^7$ kg) the probability of impact on land is approximately 10^{-12} to $10^{-13}/km^2/$ year, assuming a land surface of 130×10^6 km^2. Assuming that the total committed surface area to waste disposal is 100 km^2, the probability of an impact occurring in 100,000 years is 10^{-5} to 10^{-6}. However, the short-term effects of the impact of a meteor of the size considered here would undoubtedly be far greater than the exhumation of radioactive wastes. This same remark applies also to the detonation of a nuclear weapon at the surface of the waste disposal facility.

One is not able to reasonably estimate the probability of someone accidentally drilling through the waste disposal facility in search of minerals. For example, it is not uncommon to find oil and gas fields bearing up against salt domes. Therefore, the probability of someone drilling near a disposal facility within 10^4 years is, probably, quite large.

B2. Disposal in a Nuclear Cavity (Plowshare Concept)

A method has recently been proposed by J.J. Cohen, A.E. Lewis and R.L. Braun /56/ for the incorporation of radioactive wastes from fuel reprocessing in molten silicate rock. It involves the placement of the liquid wastes from a fuel reprocessing facility, with little or no treatment, in a deep underground cavity created by nuclear explosives. The concept simplifies the waste disposal problem in that interim storage, secondary treatment, and transportation of the wastes are eliminated. The fuel reprocessing facility would be located in the close vicinity of the cavity. The cavity would be created below a considerable thickness of impermeable rock at a depth of 2 to 3 km in silicate rock as shown conceptually in Fig. 28. High-level wastes, as well as intermediate - and low-level wastes, would be injected into the cavity. The temperature of the rock would have to be maintained below its melting point during operation of the facility by adding cooling water. For this reason two bore holes to the cavity would be required, one for waste and cooling water addition and the other for return of steam. The steam would be condensed and recirculated to cool the cavity or processed for reuse in the reprocessing facility. The entire concept is a closed system. If for some reason the flow of cooling water to the cavity is disrupted the cavity and access holes would melt. It would be, conceivably, difficult to initiate operation of the cavity again.

When the cavity has reached its capacity its access holes would be permanently sealed. The rock surrounding the cavity would then begin to melt due to the lack of cooling. The molten rock cavity would increase in size for a number of years after closure of the access holes until a heat balance is established. When the molten rock begins to cool and solidify the radioactive wastes would be incorporated into the rock.

The concept, as envisioned by Cohen et al. /56/, entails a cavity created by a 5 kt nuclear device with a useful service life of 25 years connected to a 1500 ton per year fuel reprocessing facility. After closure of the cavity a maximum melt radius of 96 m would be reached within 65 years. Therefore, the wastes from one ton of reprocessed fuel would be incor-

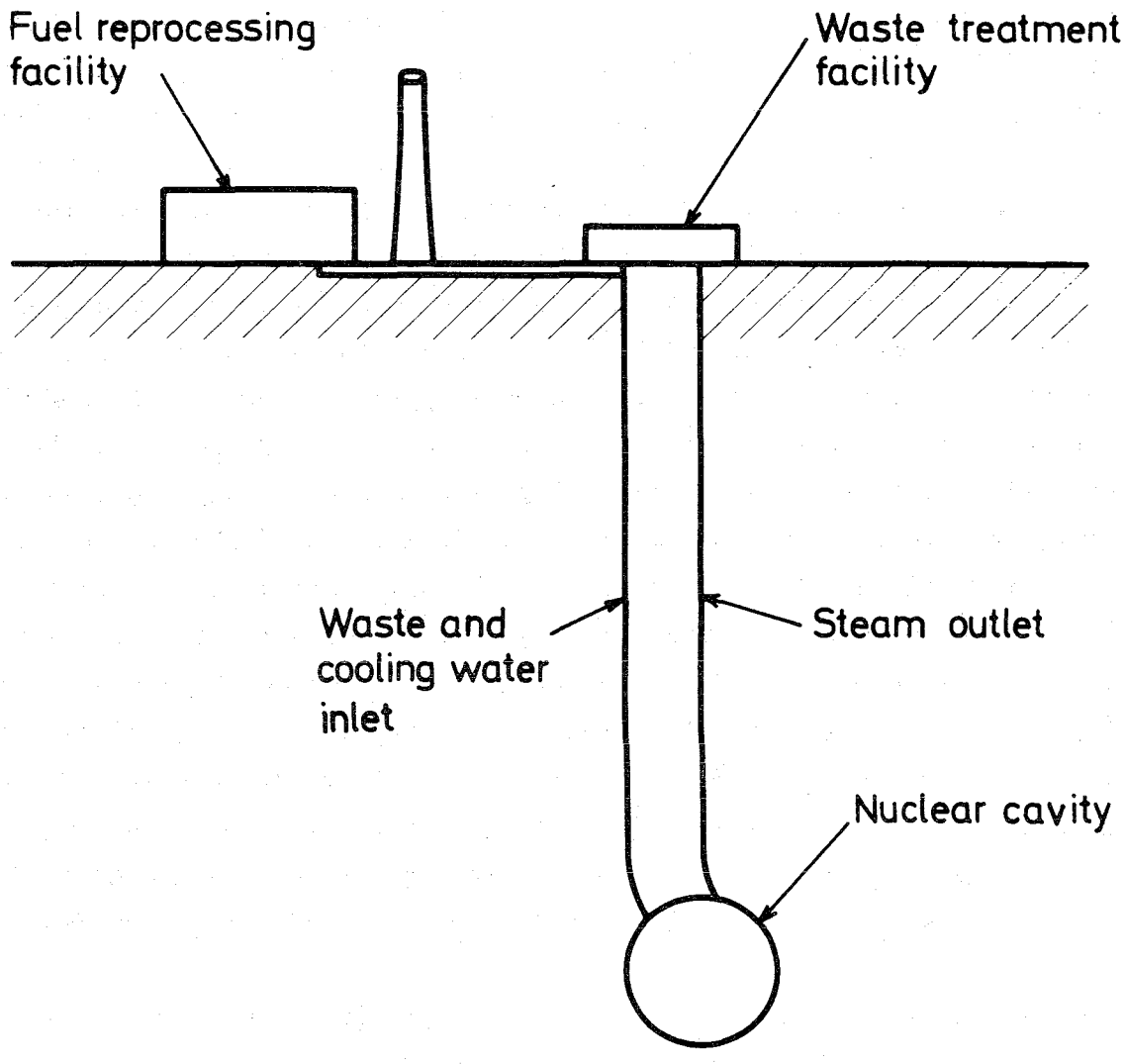


Fig. 28 Waste Disposal in a Nuclear Cavity (Plowshare Concept)

porated in about 100 m^3 of rock. Comparing this waste incorporated into the rocks with uranium ore of an equivalent amount, as done in the previous section for disposal in salt deposits, we find a hazard index of

$$HI \text{ (uranium ore)} = 0.71 \times 10^7.$$

In comparison to the salt deposits the time span in which the waste can be considered "dangerous" is about 20,000 years longer. However, in the event that something does happen the wastes in this concept are not in as favorable a form as in the salt deposits. For several hundred years after closure of the cavity the rock and wastes will remain molten. Therefore, if a large enough volume of water enters the cavity it is conceivable that a significant pressure build up could occur, forcing the molten rock and wastes out of the cavity, possibly to the surface of the earth. In other words, one could obtain a "geyser" effect. However, when most of the short half-life fission products have decayed, in about 1000 years, the remaining fission products and actinides will be incorporated in an insoluble silicate rock matrix deep underground.

In addition, because of the high temperature of the wastes in this concept the volatile isotopes will be driven out of the wastes. They, therefore, would need to be treated separately. The cooling water that circulates in the closed system would become highly tritiated (^3H : HTO) in a short time, thereby, presenting an increased radiation hazard to the personnel working in the facility, as the activity of the cooling water could reach as high as 1 Ci/m^3.

Because of the molten state of the waste and rock for a long period of time the heavier isotopes (actinides) will probably concentrate toward the bottom of the cavity. Over the estimated 25 year life of the facility approximately 37 tons of actinides will have been injected into the cavity. As a point of reference the critical mass of ^{244}Cm is of the order of 20 kg /57/ and ^{241}Am about 100 kg /58/.

B3. Disposal under an Ice Sheet (Antarctica)

Ice has several advantageous features common with salt if its average temperature is maintained well below the freezing point. Its fractures are self-healing through recrystallization or plastic flow. In addition, it is impermeable to water and has a relatively high heat conductivity compared to rock. The polar regions comprise about one-twelfth of the earth's surface and are extremely remote areas of the earth.

Zeller and Saunders /59/ have recently proposed the establishment of an international radioactive waste disposal facility in Antarctica. They suggest that the best location for such a facility would be near the "pole of unaccessibility". As suggested, the tops of ridges in the underlying bed rock are the most favorable. Here the temperature at the rock/ice interface should be lower than in the basins. This area has a land elevation of over 3000 meters above sea level and an ice thickness of 1000 m to 3000 m. The area is about 1300 km inland from the edge of the continent. This distance is a safety factor for the buried wastes, but a hazard for the transport of the wastes.

In this concept the radioactive wastes would be solidified, as in the salt deposit concept, into glass cylinders. The waste containers would be shipped annually (in January) by special ships to the Antarctica and then transported overland by sledge to the waste facility. Placed on the ice the hot cylinders would melt their own emplacement shafts to a final resting place. The emplacement shafts would self-heal within a short period. For the sizes and heat generation rates under consideration, the cylinders would sink at rates of about a meter per day. This would mean a total sinking time of 3 to 5 years for an ice depth of about 1500 m.

When the waste cylinders have reached the rock/ice interface they should remain stationary for several hundred years, due to heat generation, and not be subject to ice flow. After this period of initial heating the waste will probably move with the ice flow. As mentioned in the section on salt deposits, glaciers move on the average about 10 m/year. Under the most unfavorable conditions of straight line continuous movement of the waste they could reach the Antarctica continent edge in about 150,000 years. More important than glacier movement rates may be the presence of fluid water between the rock and ice. Such water could transport radioisotopes rather rapidly.

As evident by the heated debate in the Bulletin of the Atomic Scientists /69/, following the publication of an article by Zeller, Saunders and Angino /61/ on this subject, the possibility of an ice surge, initiated by the next ice age in 10,000 to 50,000 years or by conditions due to the hot waste cylinders, is taken seriously. Such an ice surge could transport the wastes relatively fast to the sea.

With present day knowledge it would appear that the disposal of wastes under the ice cap of the Antarctica should be limited to the short half-life fission products ($t_{1/2} \leq 50$ years). Therefore, this disposal scheme would only be a partial solution to radioactive wastes. For example, the long half-life fission products and actinides would have to be disposed by one of the other methods considered in this paper.

From a risk point of view the probability that the wastes will be unintentionally uncovered is considerably less in the Antarctica than on other continents. However, the risk due to transportation accidents is considerably greater.

B4. Disposal in the Earth's Crust by Self-Burial

The concept of self-burial of radioactive wastes has recently been proposed by Donea et. al. /62/ of Ispra, and Logan /63/ of the United States. According to this concept, high-level wastes encased in a metal sphere would be allowed to melt their own emplacement shaft in rock. The spheres would probably be released from an initial pre-drilled depth of about 2 km, and descend by gravity to greater depths utilizing the decay heat from the incapsuled radioactive wastes. Descent times and final depth achieved would depend on the heat source density of the radioactive material, the type of rock the sphere descends through, and the size, shape and material of the capsules.

The results of calculations by Logan /63/ for descent through basalt rock indicate that 1m diameter sphere containing a 30% volume fraction of fission products would come to rest at approximately 16.2 km in 105 years. The same capsule with only actinides would reach 10.2 km in 96 years, or with only the elements of strontium and cesium it penetrates the earth's crust and enters the mantle, thereby achieving depths greater than 50 km.

It would probability be necessary, for economic reasons, to use the same implantation hole for many succeeding waste capsules. The drilling cost for an emplacement hole of 2km depth is estimated at roughly one million U.S. dollars /56/.

The concept is similar to the ice sheet concept (section V. B3) in that the wastes melt their own emplacement shaft, and similar to the nuclear cavity concept (section V. B2) in that the wastes, after following the same trail, would accumulate in the same general area (therefore waste concentration). The application of this waste disposal concept would require an extension of present day technology.

The risks in this concept are initially in the handling phase of the disposal (solidification and transportation of wastes) and later result from the possible escape to the environment of material after

the capsules have been released. The waste solidification and transportation phase would not be qualitatively different than the same in the salt deposit concept (section V. B1). Both concepts entail the handling of hot capsules that must be cooled. After the capsules had been released they would be essentially irretievable, and they would probably concentrate somewhat after reaching their maximum depth. In this respect the long-term risk from this concept is similar to that in the nuclear cavity concept (section V. B2).

C. Storage

Almost any high-level waste management scheme involves an interim storage of the wastes as liquid and/or solid. The reason being that there are practical and economic advantages to be gained by allowing many fission products with short and intermediate half-lives to decay prior to additional waste processing. In addition, there is a growing element of the nuclear community that is advocating storage of radioactive wastes, not as an ultimate solution but rather as a temporary solution. The argument is that the safety of present day disposal concepts must be adequately demonstrated before disposal of the wastes is initiated, since as seen in the previous section, retrievability of the wastes can not be assumed. From storage the waste could be retrieved at any time for further treatment and management. From today's state of development the demonstration of disposal safety would require something of the order of 20 years. As a result, the concept of "Engineered Storage" /64/ has surfaced in the United States. The idea involved is to store solidified high-level wastes for a period up to 100 years in specially prepared buildings. It is assumed that within this period of time an acceptable waste disposal facilities will be able to go into operation.

For safety reasons it is assumed that any type of storage facility would be located in an area that is tectonically stable and in which the geologic materials in the surrounding vicinity have a low permeability and a high ion exchange capacity. These last criteria are advantageous to hamper the movement of any ground released activity.

C1. Storage of Liquid Wastes

For the storage of the waste as liquids in tanks, the causes of activity release would be considered to be one of the following:

1) tank corrosion
2) loss of cooling
3) hydrogen explosion
4) external causes (earthquake, sabotage, flood, etc.).

These release mechanisms are discussed below.

In a safety analysis of liquid tank storage performed at ORNL /65/ it was the opinion of the authors that the most likely mechanism of a major release of radioactive material would be loss of cooling. To date, however, tank corrosion has been the only cause of tank failure. There have also been incidents of release during filling or transferring of liquid wastes caused by operator error or plugged pipes /66/.

1) Tank Corrosion

Some insights into the problems of liquid tank storage can be gained from the experience in the United States where a very large volume (~$3.x10^5$ m^3) has been handled to data. This experience was recently reviewed by Lennemann /67/ of the USAEC. Of the 198 tanks built since 1944, 20 have leaked, all caused by either structural stress or corrosion. Some leaked shortly after they were placed in service and others after a period longer than the expected average life. A summary of these tank failures in given in the following Table 13. The data for the last tank leakage given in the table was compiled separately from Ref./68/. At the time of this last leakage the total amount of liquid in storage was about $1.6x10^5$ m^3; therefore approximately 0.3% of the total liquid waste stored was released. In only one incident is it believed that the wastes have entered ground water. Generally the wastes have remained in the surrounding soil. Although some generalizations concerning the relative degree of fixation of the principle radioisotopes in soil are possible, their behavior is so dependent on factors such as rate and amount of rainfall, drainage, etc. that general quantitative forecasts are not practical at this time.

Studies have shown that, for example, ^{90}Sr migrates at a rate of about 1.1 to 1.3 cm/day through soils that have moderately high exchange capacity and that are permeated with ground water. Since the mean life of a ^{90}Sr atom is about 40 years, the mean distance the isotope could traverse before decay would be less than 200 m under the same conditions. In addition, it has been confirmed that ^{137}Cs is more tightly bound by soil than ^{90}Sr. A general remark in this respect is that the capacity of the soils to store fission products in ionic form seems to be substantial as confirmed by measurements. The transuranium isotopes would be expected to migrate considerably further than either ^{137}Cs or ^{90}Sr, primarily because of their much longer half-lives. The annual rate of movement of these isotopes is very low due to the formation of radiocoloids. Typical mean distances of travel would be of the order of 1500 m.

Table 13 Tank Leakage Recorded by the USAEC / 67 /

Leak date	Tank Type[1)]	Years in Service	Approx. Leakage (m^3)[2)]	^{137}Cs Release (kCi)	Failure Mechanism[3)]
1956	A	9	200	0.09	SS
1957	B	2	0	0	SC
1958	A	<1	60	8	SS
1959	A	6	80	2	PC
1959	A	13	110	23	PC
1959	B	3	0	0	SC
1959	B	2	0	0	SC
1959	B	<1	<4	<0.1	SC
1960	A	7	130	4	PC
1962	A	7	10	17	SS
1963	A	<1	small		SS
1964	A	8	small		SS
1965	A	10	small		SS
1965	A	7	190	40	SS
1969	A	13	110	45	SS
1969	A	21	260	51	PC
1969	B	14	0	0	SC
1971	A	17	unknown		PC
1972	B	12	0	0	SC
1972	A	16	unknown		PC
1973	A	30	440	40	unknown

1) A-Carbon steel-lined concrete tank
B-Carbon steel tank inside partially steel-lined concrete vault

2) Average tank capacity ~3000 m^3

3) SS: structural stress
PC: pitting corrosion
SC: stress corrosion

All of the tanks that have developed leaks to date were carbon steel tanks. Since 1954 stainless steel tanks have been in use. None of these tanks have shown evidence of deterioration. In addition, the wastes in all these tanks has been alkaline. However, because of longer tank service life, significantly smaller quantities of wastes, better feed for further treatment, and cost analyses fuel reprocessing acid waste storage will be preferable. Therefore, it is not proper to deduce any conclusions concerning the future safety of liquid tank storage from the past record exhibited in Table 13 .

2) Loss of Cooling

The loss of cooling accident would be, most likely, the accident resulting in the most hazardous release of radioactive materials. The worst possible time for this accident to occur would be when the storage tank was newly filled and the heat generation rate at its peak. The heat generation rates for the wastes from the various reactor types are given in Table 14 . For our considerations here we may assume that our storage tank is located at a 1500 MT/year reprocessing plant. A representative tank capacity is 3500 m^3. At a continuous filling rate, the tank would be filled to capacity in slightly more than 2 years. Assuming 1.0 m^3 of high-level liquid waste per ton of fuel reprocessed, the average heat generation rate in the tank would be ~9 kw/m^3, or 31.5 MW total for the tank. For safety reasons these tanks would be provided with a mimimum of two independent cooling systems.

However, if for some reason the condenser and filter were out of service and the cooling systems were not able to function the contents of the tank would self-boil in a matter of a few hours. The possible causes of the loss of service of the cooling and ventilation facilities would be

1) loss of power
2) flood, hurricane, earthquake, sabotage
3) operation error or neglect.

The wastes would boil to dryness in something of the order of 100 hours, if they are acid, or roughly twice as long if they are alkaline. If the wastes were still contained at this time a temperature of more than 1000 ^{o}C would be reached in the center of the tank.

The volatile components of the wastes would be released to the atmosphere from the self-heating wastes. Because of their relatively low vapor pressure, most of the released activity would be due to cesium and rutherium. These components, together with the total fission product activity, are given in Table 15 as a function of the time since reprocessing for three different

Table 14 Heat Generation Rates for Characteristic Radioactive Wastes
(kilowatts per metric ton fuel reprocessed)

Time since Reprocessing [1] (year)	0	1	3	5	10	30
Waste from:						
PWR (Eq.Uranium)	21.0	8.2	3.2	1.8	1.1	0.6
PWR (Pu-recycle)	23.5	9.2	3.8	2.4	1.6	0.8
LMFBR (LWR-Pu)	28.2	10.4	3.1	1.5	0.8	0.5
LMFBR (FBR-Pu)	27.7	10.5	3.2	1.6	1.0	0.6
HTGR	29.9	11.2	5.2	3.5	2.3	1.3
THTR (^{233}U-recycle)	29.0	10.7	4.9	3.3	2.3	1.3

[1] Reprocessing of fuel 150 days after discharge from reactor

Table 15 Important Volatile Components in Radioactive Wastes
(curies per metric ton of fuel reprocessed)

Time since reprosessing[1] (year)	0	1	3	5	10	30
PWR (Eq. Uranium)						
^{106}Ru	4.18+5	2.10+5	5.27+4	1.33+4	4.22+2	4.29-4
^{134}Cs	2.24+5	1.60+5	8.12+4	4.13+4	7.62+3	9.02
^{137}Cs	1.10+5	1.07+5	1.02+5	9.77+4	8.71+4	5.48+4
Te(total)	1.98+4	3.81+3	1.57+3	9.39+2	2.60+2	1.53
Total fission products:	4.38+6	1.69+6	7.04+5	4.59+4	3.20+5	1.86+5
LMFBR (LWR-Plutonium)						
^{106}Ru	9.33+5	4.68+5	1.18+5	2.96+4	9.41+2	9.57-4
^{134}Cs	1.90+4	1.36+4	6.90+3	3.51+3	6.47+2	7.48-1
^{137}Cs	1.15+5	1.13+5	1.08+5	1.03+5	9.15+4	5.77+4
Te(total)	3.91+4	9.35+3	4.14+3	2.47+3	6.82+2	4.02
Total fission products:	6.32+6	2.29+6	8.32+5	4.71+5	2.87+5	1.61+5
HTGR (no ^{233}U-recycle)						
^{106}Ru	9.26+4	4.65+4	1.17+4	2.94+3	9.35+1	9.51-5
^{134}Cs	4.69+5	3.34+5	1.70+5	8.64+4	1.59+4	1.89+1
^{137}Cs	2.07+5	2.02+5	1.93+5	1.84+5	1.64+5	1.03+5
Te(total)	4.53+4	7.47+3	2.65+3	1.56+3	4.33+2	2.55
Total fission products:	6.34+6	2.42+6	1.22+6	9.08+5	6.84+5	3.99+5

[1] Reprocessing of fuel 150 days after discharge from reactor

waste streams. Assuming $1m^3$ of liquid waste for each ton of fuel reprocessed, the figures in Table 15 would be than per m^3 of tank waste.

Blomeke and Emerson /65/ estimate that during the self-boiling phase of the accident, without the condenser and filter in service, something of the order of 300 Curies of rutherium and 10^3 curies of other fission products would be released per minute, assuming acid wastes. Over a period of 175 hours 90% of the rutherium, cesium, and tellurium as well as 5% of other fission products will have been released. In our case this would represent a release to the atmosphere of about 1.9×10^9 curies. For alkaline wastes these figures would be considerably reduced, of an order of magnitude.

To properly analyze this accident within the framework of risk it will be necessary to calculate the probability and time, after reprocessing, of the accident occurring. This could be handled with a fault tree analysis of the tank storage facilities.

3) Hydrogen Explosion

The intensely radioactive Purex concentrates cause radiolytic decomposition of H_2O. The rate of production of hydrogen is such that if the ventilation system failed, the lower explosive limit of 4% hydrogen in air would be reached in a matter of hours. Sufficient quantities could accumulate which could explode with a force sufficient to rupture the tank and concrete encasement. This would probably cause, additionally, a loss of coolant for the wastes. If the tank was so ruptured that the wastes were still contained the accident would proceed as in the pure loss of coolant accident. However effective remedial action would be more difficult in this case. The more likely condition is that the tank would rupture such that some or much of the contents would seep out into the soil. This would greatly reduce the impact of the accident.

4) External Causes

There are numerous externally created incidents which could lead to releases of the wastes from the tank. The most important of these are the events created by severe weather conditions, such as floods, hurricanes, tornadoes, etc. In addition, there is always the possibility of sabotage. The most susceptible portion of the storage facility would be the surface located cooling and ventilation systems. In this case the accident would proceed as illustrated in the loss of cooling accident explained above. However, if the external causes were to rupture the tank directly the wastes would seep into the ground.

C2. Storage of Solid Wastes

The possible causes of tank failure given in the previous section for storage of liquid wastes are applicable to the storage of solid wastes except that no radiolytic hydrogen formation is possible, therefore the possibility of a hydrogen explosion does not exist.

The leakage of activity from containers holding solid wastes could be caused by defective sealing, by release of overpressure built up inside the container, or by abnormally high corrosion rates. In any case the leakage rate would be slow from the solid, and could be detected by monitors so that remedial action could be taken.

The loss of coolant accident would probably be the most severe accident in a solid waste storage facility, as in the case of liquid storage. However, since the wastes would be at least several years older than the liquid wastes, due to the time lag in solidification after reprocessing, the heat generation rates per unit volume would be correspondly smaller. In addition, the waste may be combined in the solidification process with an amount of inert material. Therefore, the likelihood that the waste could reach the melting point of stainless steel is small. If the waste container does fail, by overheating or by rupturing from collapse of the surrounding structure, the volatile components of the wastes would be released to the atmosphere from the molten wastes. Again the released activity would be due primarily to cesium and rutherium.

Release rates of volatile elements from ceramic wastes have been measured to be of the order of 0.5 to 1.0% of the content per hour. A typical solid waste container would contain, roughly, the wastes from one to three tons of reprocessed fuel. Assuming the wastes have aged five years before they are solidified, our typical waste container would

contain approximately 3.0×10^5 curies of cesium and rutherium. Therefore, the release from one container would be of the order of 3000 curies/hour. For younger wastes the release rate would be correspondingly higher.

The rate at which the loss of coolant accident would progress depends on the mode of cooling. In an air cooled facility the accident would progress much more rapidly than in a water cooled facility because of the absence of water to evaporate. It is also likely that in an air cooled facility the waste containers would be smaller and spaced at larger distance. Therefore, the atmospheric release would be somewhat smaller.

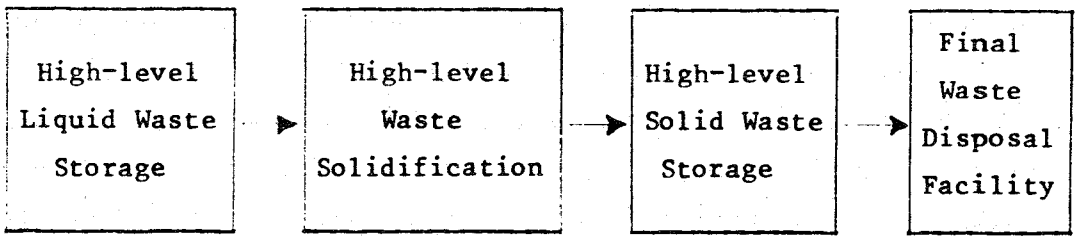
The nonvolatile components of the wastes would, in any event, remain in the storage facility. If the floor of the storage facility was ruptured they would seep into the soil and be subject to leaching by water.

VI. Evaluation of Risks from Waste Management and Disposal

Normally at this stage one would expect that a combination of the methods illustrated in section III. of this report with the data presented in sections IV. and V. would yield the sought after values. However, this is presently not possible due to the lack of sufficient data from a complete detailed analysis of the various waste disposal schemes. In addition, as obvious from the remarks in section V. there appears to be no one single optimum waste disposal scheme, but rather one would believe that an optimum strategy would consist of a combination of two or more of the schemes. We should not overlook the fact that in a consideration of the various waste disposal schemes, a particular scheme may require the support of other waste treatment and handling activities. The risks due to the support activities must be added to the risk due to the disposal scheme itself. For example, in the salt deposit concept the waste will be stored for a period as a liquid, be solidified, stored again as a solid before being deposited in the salt. Each of these activities have a risk associated with it. In the plowshare concept these activities are not necessary, therefore to leave them out of the overall consideration of the risk from disposal in salt deposits would be to neglect a component of its risk.

Realizing that it is not possible at present to work forward in risk analysis of waste disposal schemes, as outline in the preceding sections, one could work backwards. In other words, by setting a particular risk standard one could search for the necessary criteria to be maintained to meet the risk standard. These criteria would be decontamination factors in waste processing facilities, and upper permissible limits on accident probabilities. To illustrate this approach we shall treat the problem of disposing of solidified wastes in a salt deposit. We shall assume a nominal spent fuel throughput of 1500 t/yr. The wastes

from the reprocessing of these 1500 t of fuel per year will be stored 5 years as a liquid, solidified, and stored an additional 5 years before being brought into the salt deposit. In addition, the average annual aeolian dilution (X/Q) is 10^{-7} sec/m^3 for air and for groundwater (X/Q) is 10^{-1} sec/m^3. The chain of activities we shall consider is illustrated by the following diagram. The risk from both normal opera-



tional releases, if there are any, and accident conditions are considered. The risk standard, given as a maximum offsite dose, can be arbitrarily set at 30 mrem/yr. for the normal operational releases from any one facility and also for the expected accident risk from each facility.

Normal Operational Releases

The only square in the flow diagram above that would have normal operational releases is the waste solidification facility. There are at present several different processes under investigation for the solidification of the high-level wastes. Generally the semi-volatiles such as ruthenium, cesium, selenium, tellurium and technetium can be off-gased because of the high temperatures used. It is possible, in some cases,

to add substances to the waste which form thermally stable components with cesium. The release of ruthenium may be reduced by denitration of the waste solution.

As a function of the decontamination factor the maximum average annual offsite doses, based on a conservative assumption that 1/10 of the ICRP recommended maximum permissible concentration in air (with the lung as the critical organ) are given in the following table. It is worth mentioning that a decontamination factor of 2000 is reported for the cal-

Table 16 Estimated Offsite Doses from the Semi-Volatiles Released During Waste Solidification (mrem/yr.)

Decontamination Factor	PWR Fuel	LMFBR Fuel
2000	958	2820
10^5	4.79	14.1
10^6	0.48	1.41

cining process at the Idaho Chemical Processing Facility. Therefore, for a throughput of wastes from 1500 t/yr. of spent fuel, assumed here, a significant improvement in the decontamination factor is required to meet our criterion of 30 mrem/yr. A required decontamination factor would be of the order of 10^5.

Accident Conditions

1) Liquid Waste Storage - Assuming that the liquid wastes are stored for 5 years between fuel reprocessing and waste solidification, the total waste quantity in storage at any given time for a 1500 t/yr. throughput would be

$$1500 \text{ t/yr.} \times 1 \text{ WU/t} \times 5.0 \text{ yr.} = 7500 \text{ WU}$$

where WU = one waste unit and represents the total quantity of high-level wastes from the reprocessing of one ton of spent fuel.

For the upper permissible accident limit we shall assume that all of the semi-volatile radionuclides and 5% of the remaining fission products are released to the atmosphere. The accident is assumed to take place at the midpoint of the total liquid waste storage period (2.5 years). Rather than calculate separately for the PWR and LMFBR case, we shall use only the larger risk value, per unit waste, in the following considerations. At 2.5 years after fuel reprocessing the added sum of the ratio of curies to the corresponding MPC values for the radionuclides released in the assumed accident is

$$\sum_i \frac{Q^i}{MPC_a^i} = 1.48 \times 10^{15} \text{ m}_a^3/\text{WU}$$

The calculation of the upper limit accident probability, to yield an expected risk of 30 mrem/yr., is as follows:

$$P\left[1/\text{sec}\right] * 1.48 \times 10^{15} \left[m_a^3/\text{WU}\right] * 7500 \left[\text{WU}\right] * 10^{-7} \left[\frac{\text{sec}}{m_a^3}\right] * 500 \left[\frac{\text{mrem}}{\text{yr}}\right] \leq 30 \left[\frac{\text{mrem}}{\text{yr}}\right].$$

Solving for P

$$P\left[1/\text{sec}\right] \leq 5.4 \times 10^{-14} \quad \text{sec}^{-1}$$

or

$$P\left[1/\text{year}\right] \leq 1.7 \times 10^{-6} \quad \text{yr.}^{-1} .$$

Assuming a tank capacity of 3000 WU, the accident probability per tank per year is

$$P \leq 4.3 \times 10^{-5} \quad \text{yr.}^{-1} \quad \text{tank}^{-1} .$$

2) Waste Solidification Facility - As mentioned previously, the waste solidification is assumed to take place 5 years after fuel reprocessing. Our upper limit accident is assumed to be a total release to the atmosphere of all the semi-volatiles radionuclides. In addition we assume that 1/200 of the annual throughput would be involved in the accident, i.e. 1/200*1500 WU = 7.5 WU. At 5 years after fuel reprocessing

$$\sum_i \frac{Q^i}{MPC_a^i} = 5.91 \times 10^{14} \; m_a^3/\text{WU}$$

for the semi-volatiles in the waste. The upper accident probability is calculated as follows:

$$P\left[1/\text{sec}\right] * 5.91 \times 10^{14} \left[m_a^3/\text{WU}\right] * 7.5 \left[\text{WU}\right] * 10^{-7} \left[\frac{\text{sec}}{m_a^3}\right] * 500 \left[\frac{\text{mrem}}{\text{yr}}\right] \leq 30 \left[\frac{\text{mrem}}{\text{yr}}\right]$$

$$P\left[1/\text{sec}\right] \leq 1.38 \times 10^{-10} \quad \text{sec}^{-1}$$

$$P \leq 4.3 \times 10^{-3} \quad \text{yr.}^{-1} .$$

3) Solid Waste Storage - Solid waste storage is assumed to be for a period of 5 years after solidification and before final disposal. Therefore a total capacity needed for a facility would be for the wastes from 5 years of reprocessing 1500 t/yr. of spent fuel. Again, for the upper limit accident we assume a total release to the atmosphere of all the semi-volatiles present in the wastes and that the accident would occur at the midpoint of the total storage time (7.5 years after fuel reprocessing). In this case our ratio curies to MPC values is, per volume waste unit (WU)

$$\sum_i \frac{Q^i}{MPC_a^i} = 5.5 \times 10^{14} \left[m_a^3/\text{WU} \right]$$

and the accident probability

$$P\left[1/\text{sec}\right] * 5.5\times10^{14}\left[m_a^3/WU\right] * 7500\left[WU\right] * 10^{-7}\left[\frac{\text{sec}}{m_a^3}\right] * 500\left[\frac{\text{mrem}}{\text{yr}}\right] \leq 30\left[\frac{\text{mrem}}{\text{yr}}\right]$$

$$P\left[1/\text{sec}\right] \leq 1.45 \times 10^{-13} \text{ sec}^{-1}$$

$$P \leq 4.6 \times 10^{-6} \text{ yr.}^{-1} .$$

Assuming that an individual waste cylinder contains the wastes from 2.5 t of reprocessed fuel, the accident probability per year per cylinder is

$$P \leq 1.4 \times 10^{-3} \text{ yr.}^{-1} \text{ cylinder}^{-1} .$$

To summarize, the permissible upper limits of the accident probabilities, assuming an expected accident risk of 30 mrem/yr., are collected in Table 17.

Table 17 Permissible Upper Limit of Accident Probabilities

Facility	Probability
Liquid Waste Storage	4.3×10^{-5} yr.$^{-1}$ tank^{-1}
Waste Solidification	4.3×10^{-3} yr.$^{-1}$
Solid Waste Storage	1.4×10^{-3} yr.$^{-1}$ cylinder^{-1}

From these considerations it appears that the most stringent safety requirements have to be meant in the liquid waste storage. As mentioned in section V.C1. one has to engineer against the permanent loss-of-cooling accident.

Final Waste Disposal Facility Accident

As we have done previously we can also attempt here to find an upper limit of the accident probability which would give us an expected accident risk of 30 mrem/yr. This is much more complicated here because of the time dependence of the hazards of the waste (see section IV.) and the continuous accumulation of the waste in the facility. However, one would expect that if the wastes were added to the facility at a constant annual rate an equilibrium value of the hazard index would be ultimately reached. The equilibrium value would only be a function of the annual rate of addition and the hazard index of this amount. The approach to an equilibrium value is shown in Fig. 29 for the case of a constant yearly addition of a quantity of waste having an initial hazard index of HI = 4.14×10^{11} m_w^3. The equilibrium value is reached in 800 years and is roughly 21 times the constant yearly addition. However, 95% of the equilibrium value is achieved in only 190 years.

The advantage of using the equilibrium value in our calculations is that the time dependence drops out of our problem and that it is not necessary to consider the total quantity of wastes stored. The permissible limit of the accident probability so calculated, for the equilibrium value, would be then the true upper limit.

The hazard index of the waste composition at 10 years after fuel reprocessing is, as defined by Eq. (2) in section IV.,

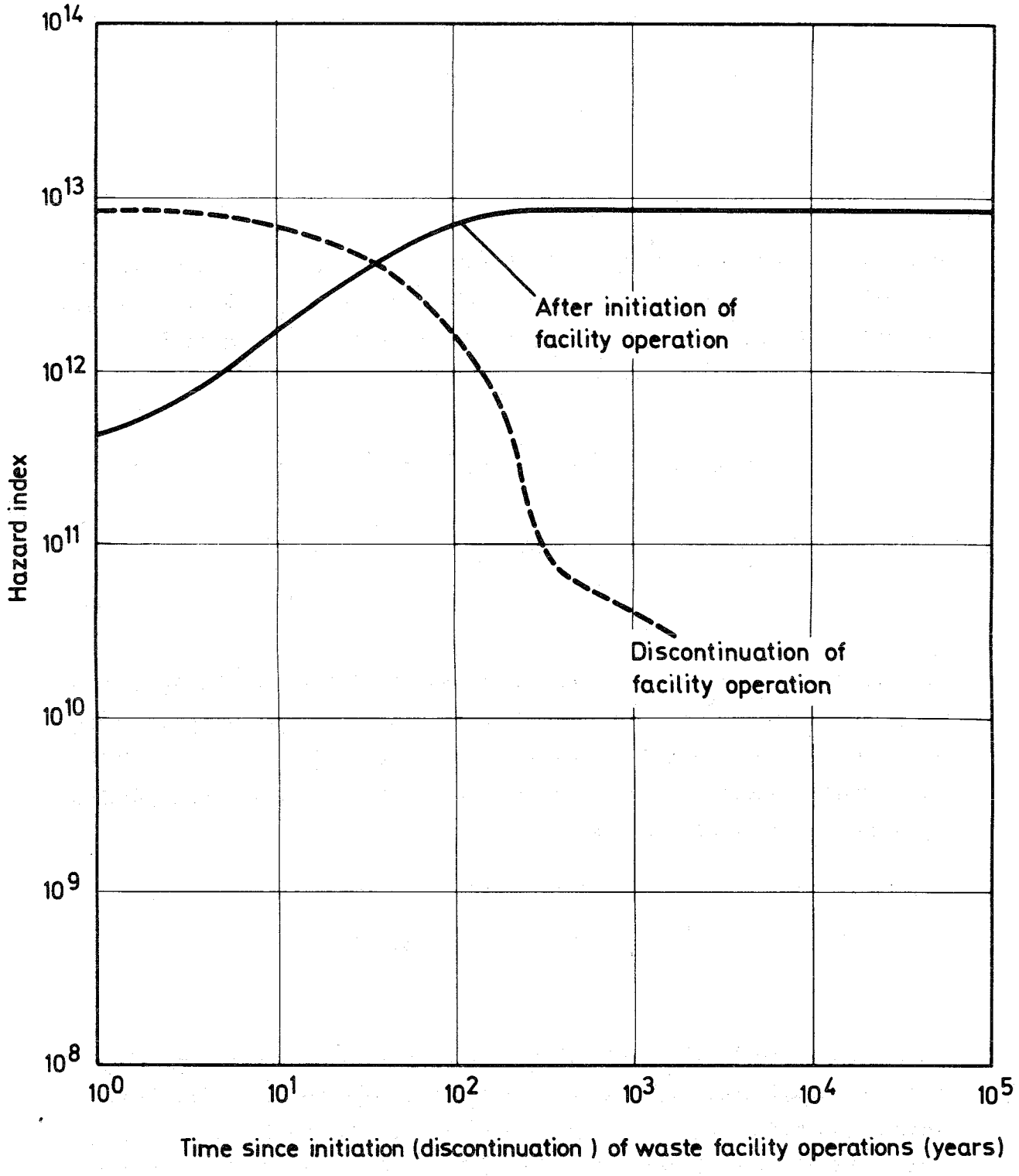


Fig. 29 Hazard Index of Total Waste Storage Facility at a Constant Yearly Rate ($Hi(t) = 4.14 \times 10^{11}\ m^3$) of Addition

$$HI = \sum_i \frac{Q^i}{MPC_w^i} = 2.13 \times 10^{11} \ m_w^3/WU.$$

With our assumed throughput of 1500 t/yr. of spent fuel reprocessing the annual addition rate to the disposal facility would be 1500 WU/yr. Therefore the equilibrium hazard index achieved in this case is

$$HI_{eg} = 2.13 \times 10^{11} \ m_w^3/WU * 1500 \ WU * 21$$

$$= 6.75 \times 10^{15} \ m_w^3 .$$

The leach rate measured for various types of solidified wastes ranges typically from 10^{-4} to 10^{-7} gm/cm^2/day. If we utilize the upper value, 10^{-4} gm/cm^2/day, and assume the waste cylinders to be 20 cm in diameter with a density of 3.0 gm/cm^3, the fraction of the waste leached by water is calculated to be

$$f = 0.77 \times 10^{-10} \ sec^{-1}.$$

It is unreasonable to assume that if an accident does occur in the waste disposal facility that all waste cylinders would be simultaneously exposed to a leaching action of water. One would suspect that the probability a certain fraction of the waste cylinders be exposed to water in an accident would exhibit a behaviour, on a log-log scale, as shown in Fig. 30.

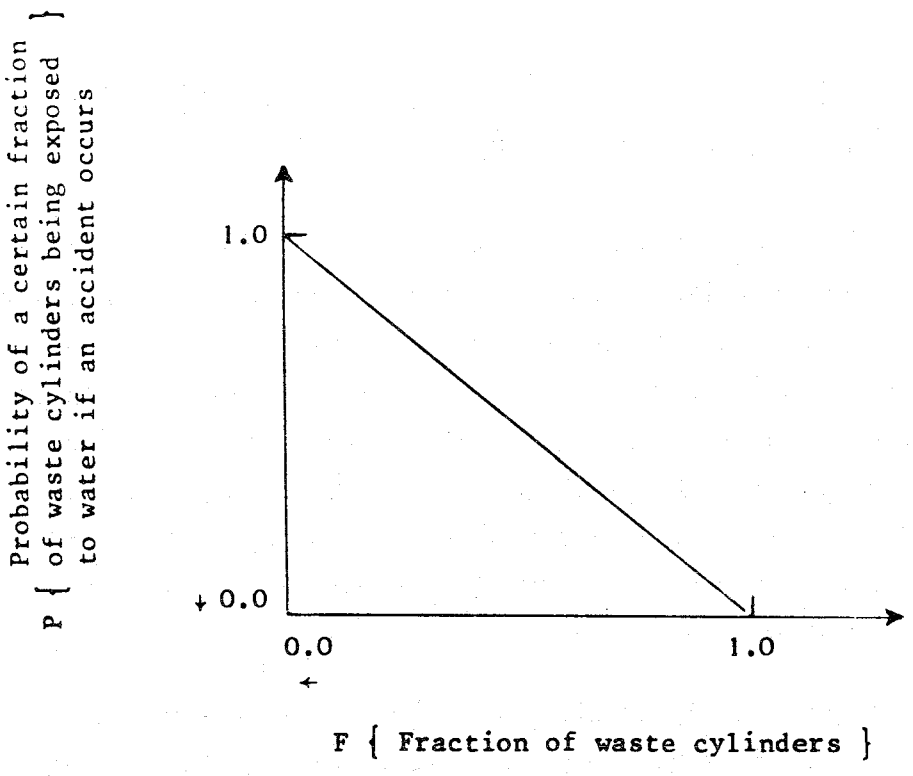


Fig. 30 Probability vs. Fraction of Waste Exposed in an Accident

In the following calculation we shall use the product P*F as the parameter, assuming an <u>expected</u> risk of 30 mrem/yr. Therefore

$$P*F*6.75\times10^{15}\left[m^3\right]*0.77\times10^{-10}\left[sec^{-1}\right]*10^{-1}\left[\frac{sec}{m^3}\right]*500\left[\frac{mrem}{yr}\right]\leq 30\left[\frac{mrem}{yr}\right]$$

where we have assumed (X/Q) of $10^{-1}\left[sec/m^3\right]$ for ground water and have neglected any filter action of the ground soil solving for P*F

$$P*F \leq 1.15 \times 10^{-6}.$$

VII. Systems Approach to Waste Management (Suggestions and Directions for Continued Effort)

As noted throughout the discussions in the preceding sections of this paper, each particular waste disposal concept has associated with it problems (or hazards) which would not have been encountered if that particular concept was not utilized. In addition, it appears that there is no one single optimum disposal concept. Rather a combination of two or more concepts may be ultimately optimum. This leads one naturally to the conclusion that a proper consideration of the overall problem of risk evaluation, and reduction or optimization, must be considered from a system point-of-view.

In addition, as emphasized many times in the preceding sections, this paper is not intended to present solutions to the problem of radioactive waste management. Rather it is to serve to point out the important points in the field and the areas in which effort is needed to solve some of the many still existing problems. Several years of concentrated effort will be needed to resolve the primary issues of the problem as outlined in the following.

In the following we present and outline the suggestions for continued effort in this field in the form of a system model approach to the problem.

System Model

A model for the overall consideration of risk evaluation for nuclear energy is given in Fig. 31. In the sense that risk is used in the following we shall always be referring to total risk due to nuclear energy, i.e. from nuclear reactors, fuel reprocessing facilities, as well as from waste treatment and disposal facilities. The individual boxes in the flow diagram of Fig. 31 are discussed below.

1) Energy Needs: The driving force of our entire system is, of course, the energy demands of the segment of the population

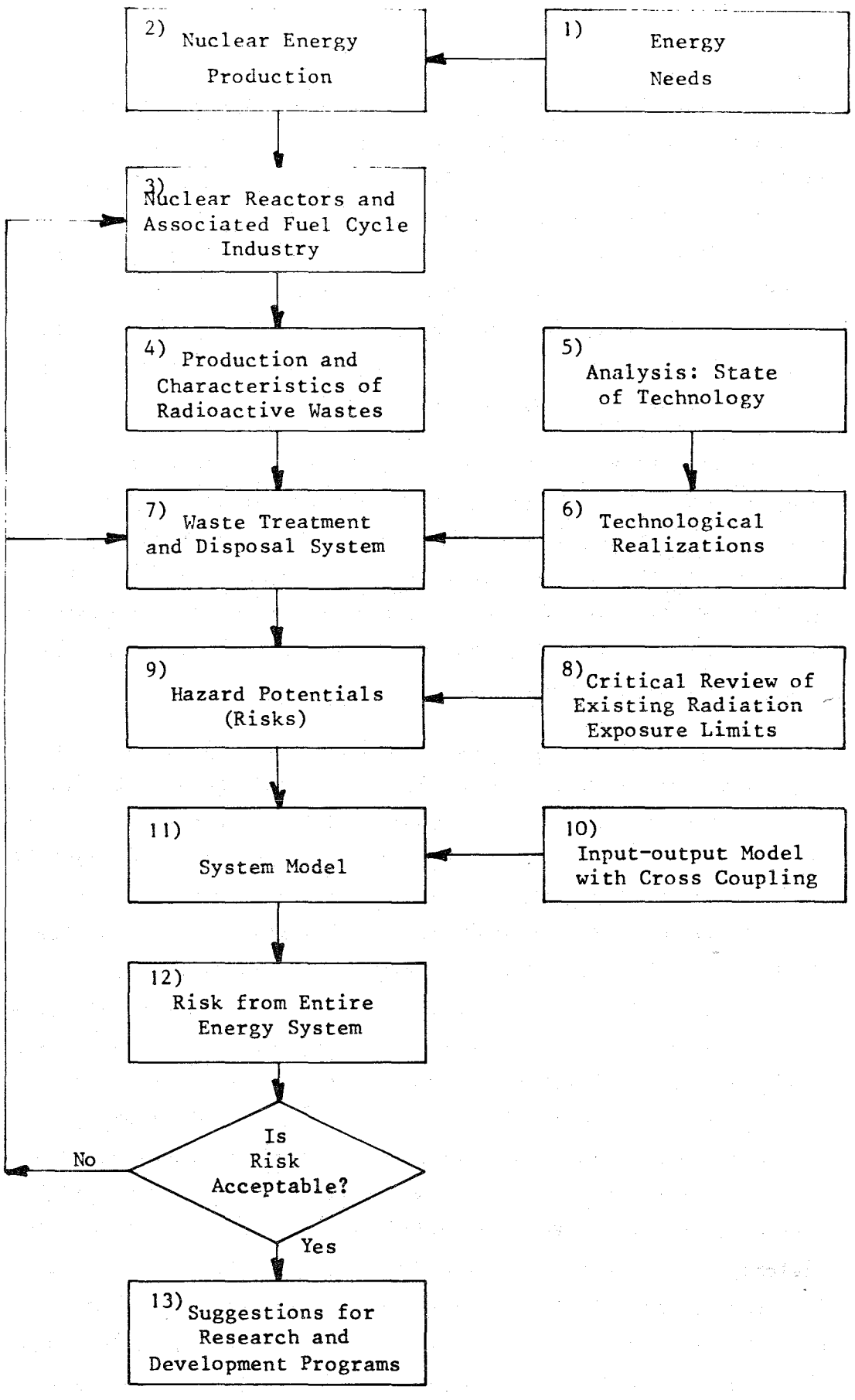


Fig. 31 Systems Approach to Evaluation of Nuclear Fuel Cycle Risks

being considered in our problem. Accurate predictions of future energy demands is essentially impossible (see discussions in section II B and Ref. /3/). However, global trends for our problem may be sufficient, as done in Ref. /7/, an input for this study. Nevertheless, the inputs here should be continually examined to check their reasonability. It is important to consider here the relative contributions of the total energy production to electrical, residential and commercial, industrial, and transportation requirements.

2) Nuclear Energy Production: The portion of the total energy needs (demands) produced by nuclear energy are estimated. In some respects the feasibility of technological developments must be assumed or assessed, as for example the application of high-temperature gas cooled reactors for process heat production, that is, use of nuclear energy aside from only electrical energy production.

3) Nuclear Reactors and Associated Fuel Cycle Industry: From the total nuclear energy production it is necessary to assess the distribution of reactor types among the total required to cover the demands. Again the feasibility of technological developments is important. For example, the possible introduction of the carbide fast breeder reactor could be important because of its lower plutonium inventory. In addition to the distribution of the reactor types, the corresponding unit sizes, their respective sites, and environmental protection equipment is necessary. For example, the requirement to scrub iodine from the off-gas effluents would not only affect the overall plant efficiency, thereby reducing the number of kilowatt hours produced per unit "risk", but it would result in a concentration of the material which in itself may be more hazardous than the simple continuous release of the material to the atmosphere. This is also true for the effects of reactor sites in relation to the average distance to energy use. Increased transmission losses increase the amount of radioactive material produced per kilowatt-hour actually "used".

In addition to the reactors, the site and size of the fuel fabrication and reprocessing facilities are also important. The transportation of fuel, both fresh and spent, must be also considered.

4) Production and Characteristics of Radioactive Wastes: Even today it is difficult to generalize about the production and characteristics of wastes produced by reactor operation since the reactors are not off-the-shelve products. In addition, the characteristics of wastes produced in fuel reprocessing facilities are to some extent unknown. For example, in the BRD there is presently only a small experimental fuel reprocessing facility, designed to accept low burn up fuel, in operation. There is also not a great deal of experience in the large scale production of plutonium fuel elements, which will undoubtedly represent a significant portion of the fuel element fabrication in the future. Efforts in this area will entail a continuing effort of data collection and appraisal.

5) Analysis-State of Technology: Before any concept of a waste treatment and disposal facility can be appraised it is necessary to assess the state of the art of our technological capabilities, as applicable to the various proposed concepts. For example, the state of the art in the production of solidified wastes, and their characteristics, is of primary importance. To mention a few others, the present day ability to decontaminate a waste stream and cleanly separate particular fractions of the waste stream is of interest in all concepts.

6) Technological Realizations: Again we come into an area requiring a certain degree of prediction capability on our part. The accuracy with which the tasks in this area can be accomplished depends directly on how well we did our homework in the preceding task, number 5. What is needed here is a prediction of our ability to meet certain required technological states of development, for the various proposed waste management schemes, at the required time. The required time is a function of the predicted accumulation of the wastes (task 3).

7) Waste Treatment and Disposal System: In essence this segment represents a refinement of what we have done in section V of this paper. That is, the characteristics of the various proposed waste management schemes must be carefully classified. Possible weak points in the schemes must be identified as well as events which could lead to subsequent releases of radioactive materials. The time periods in which the events could occur is important for the determination of the relative hazards of the various components of the waste. For example, the actinides are not readily reconcentrated by organisms, nor do they move fast through food chains, in contrast to some fission product elements. An attempt to assess the probability of occurrence of the possible events leading to radioactive material releases is also necessary. Also important for an optimum selection of a scheme, or schemes, is the associated cost for the scheme.

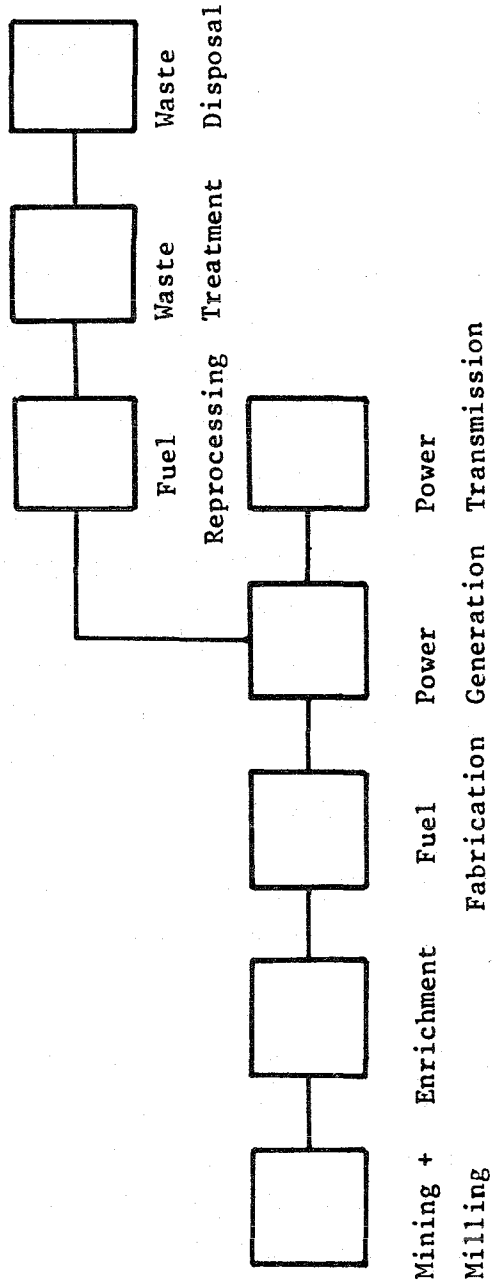
8) Critical Review of Existing Radiation Exposure Limits: As pointed out in section IV of this paper the present day radiation exposure limits, or for that matter the permissible concentrations of radioactive material in liquid and gaseous effluents of a facility, are not directly applicable for our problem. It is of paramount importance to consider the relative ease with which the various isotopes move through our environment and the extent to which they are reconcentrated by organisms. These factors must be considered in evaluating existing radiation standards for use in the assessment of waste management schemes.

9) Hazard Potentials (Risk): A combination of the input data from tasks 7 and 8 leads to an assessment of the hazard potentials of the various segments of each particular waste management concept.

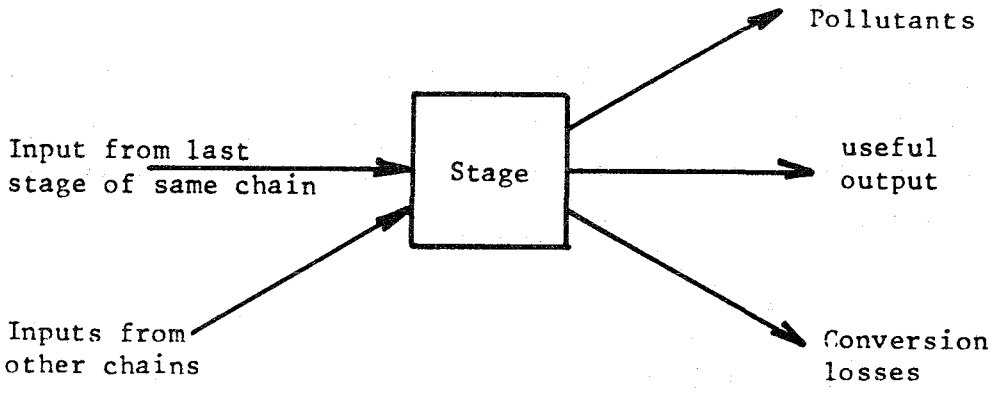
10) Input-Output Model with Cross Coupling: To rationally consider an energy system composed of multiple stages, and interconnected to other energy systems, it is necessary to develop a mathematical formulation for the system. A very

easy means of doing this is by application of the "input-output" model approach common in the field of economic and operational research. Such a model for the evaluation of the environment impact of energy systems has been developed by Maxim and Brazie /69/.

The idea behind the model is that a particular energy system is considered as a chain consisting of a series of stages (activities). The nuclear energy chain would look, very simply, as the following:



Each particular stage of the energy chain would be represented as depicted in the diagram below.



All the inputs and outputs of the stage can be expressed as relations of the primary input to the stage (see Ref. /69/). For the purpose of evaluating risk we would need to slightly extend the model by considering the damaging effects of the pollutants released. The importance of the release rates of the pollutants, the location of the release (site of the facility), the climatic conditions, etc. are not to be overlooked. With this type of representation of the energy system the overall effect of a particular "environmental protection" measure can be evaluated. It is possible that the reduction of risk (pollutant release) in one stage may elevate the total risk of the energy system.

11) System Model: The input data, from task 9, are utilized in the system model developed in task 10 to describe the total energy risks.

12) Risk from Entire Energy System: The efforts in this task involve an interpretation of the results from all the above tasks. At this point an attempt is made to assess whether the estimated risk is acceptable, not only to the system analyst but also to the "public", whoever they may be.

In the event that it is assessed that the risks are not acceptable then the various segments which could be altered for risk reduction must be identified. For example, it may be possible to alter various portions of the waste treatment and disposal systems to achieve the necessary risk reduction. At the very highest level constraints on certain portions of the fuel cycle activities could be levied.

13) Suggestions for Research and Development Programs: Based on what constitutes the final selected full nuclear fuel cycle system with associated waste treatment and disposal schemes, the necessary R + D work to achieve the proposed concepts would be outlined.

VIII. Conclusions

From what is written in the introduction of this paper, there should be no "conclusion" for this paper. In the true sense of identifying an optimum waste management stratety and its associated hazards (risks) there is no conclusion. However, in the sense of the problem there are many conclusions. Offhand it appears that there will not be a single optimum waste management strategy, but rather a combination of strategies. The technology for the realizations of the strategies is, for the most part, still to be achieved and demonstrated.

Many of the inputs for a total risk evaluation still need to be identified and quantified, and what is presently available needs to be refined considerably. A combination of methods from the area of the physical sciences and Operations Research appear to suffice for the purpose of seeking an optimum solution to the problem of radioactive waste management.

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References

/1/ J.A. McBride, "The End of the Cycle: Transportation, reprocessing and waste management - an overview", presented at the Atomic Industrial Forum workshop on The End of the Cycle, Chicago, Illinois, August 22-25, 1971.

/2/ U.S. Atomic Energy Commission, Directorate of Licensing, Fuels and Materials, "Environmental Survey of the Nuclear Fuel Cycle", Washington, D.C. (1972).

/3/ "Summary and Conclusions of Report on Future Energy Consumption Predictions", Atomic Energy Clearing House 18:39, p. 2, Sept. 25, 1972.

/4/ "Current Status and Future Technical and Economic Potential of Light Water Reactors", WASH-1082, March, 1968.

/5/ H. Mandel, "Strukturen der Nuklearen Stromerzeugung in den 70er und 80er Jahren", Atomwirtschaft, p. 18, Januar, 1973.

/6/ W.J. Schmidt-Küster, "Stand und Aussichten der Kernkraftwerke-Energewirtschaft, Industriepolitik , staatliche Förderung", Jahrbuch der Atomwirtschaft 1973.

/7/ P.E. McGrath, R. Schröder, H. Haug, "Eine Prognose über die Entwicklung der Kernenergiewirtschaft in der BRD bis zum Jahre 2010", report in preparation.

/8/ Liquid Metal Fast Breeder Reactor Demonstration Plant, Environmental Statement, USAEC, WASH-1509 (1972).

/9/ W. Vinck et. al., "The Present and Future Situation of Nuclear Energy", in Colloque International l'Energie Nucleaire et l'Environment", Liège, 22-25 janvier 1973.

/10/ "Final Environmental Report for Fort St. Vain Nuclear Generating Station", USAEC. DOCKET No. 50267-51 (1972).

/11/ P.E. McGrath et al., "Construction of a generalized fuel cycle program system and associate experience", paper in preparation.

/12/ M.J. Bell, "ORIGEN - The ORNL Isotope Generation and Depletion Code", ORNL - 4628 (1973).

/13/ P. Schmiedel and Kollmar, Kraftwerk Union, Erlangen, personal communication (1973).

/14/ R. Schröder, Projekt Schneller Brüter, Kernfroschungszentrum Karlsruhe, personal communication (1973).

/15/ Oak Ridge National Laboratory. "An Evaluation of High-Temperature Gas-Cooled Reactors", USAEC, WASH-1085, (1969).

/16/ Chauncey Starr, "Benefit-Cost Relationships in Socio-Technical Systems", Proc. Symposium on Environmental Aspects of Nuclear Power Stations", New York, 14 August, 1970.

/17/ H.F. Otway, "The Application of Risk Allocation to Reactor Siting and Design", LA-4316, 1970.

/18/ J. Weisman, "Uncertainty and Risk in Nuclear Power Plant Design", Nuclear Engineering and Design, 21, 396-405, (1972).

/19/ H.J. Otway, R.K. Lohrding and M.E. Battat, "A Risk Estimate for an Urban-Sited Reactor", Nuclear Technology, 12, 173 (1971).

/20/ G Weber, "Einführung in Methoden und Probleme der Zuverlässigkeit", KFK-1811 (1973).

/21/ D.W. Miller and M.K. Starr, The Structure of Human Decisions, Prentice-Hall, Inc., New Jersey, (1967).

/22/ J. von Neumann and O. Morgenstern, Theory of Games and Economic Behavior, Princeton University Press, Princeton, New Jersey (1944).

/23/ L.J. Savage, "The Theory of Statistical Decision", J. American Statistical Association, 46, 55-67 (1951).

/24/ H. Raiffa, Decision Analysis, Addison-Wesley, Reading, Massachusetts (1968).

/25/ M.S. Feldstein, "The Derivation of Social Time Preference Rates", Kuklos 18, 277-287 (1965).

/26/ Life Insurance Fact Book 1973, Institute of Life Insurance New York, New York (1973).

/27/ E. Turban and M.L. Metersky, "Utility Theory Applied to Multivariable System Effectiveness Evaluation", Management Science, 17, (B), 817-828, (1971).

/28/ R.L. Keeney, "A Decision Analysis with Multiple Objectives: The Mexico City Airport", The Bell Journal of Economics and Management Science", 4, 101-117 (1973).

/29/ M. Eisenbud, Environmental Radioactivity, Second Edition, Academic Press, New York (1973).

/30/ H.M. Parker, "Exposure Pathways", in Public Health Implication of Radioactive Waste Releases by C.P. Straub, World Health Organization (1970).

/31/ F. Gera and D.G. Jacobs, "Hazard Potential of Radioactive Wastes", in proc. of international conference on Radioecology Applied to the Protection of Man and his Environment, EUR-4800, Rome, Sept. 1971.

/32/ M.V. Gregory and M. Steinberg, "A Nuclear Transmutation System for the Disposal of Long-Lived Fission Product Waste in an Expanding Nuclear Power Economy", BNL-11915 (1967).

/33/ V.A. Davidenko, "On Electronuclear Breeding", Soviet Atomic Energy, 29, 866-69, July 1970.

/34/ A.M. Weinberg, "How Can Man Live with Fission", IIASA Planning Conference on Energy Systems, Laxenburg, Austria, July 1973.

/35/ W.C. Wolkenhauer, "The Controlled Thermonuclear Reactor as a Fission Product Burner", Trans. Am. Nucl. Soc., 15, 92-93 (1972).

/36/ M. Steinberg, G. Wotzak, and B. Manowitz, "Neutron Burning of Long-Lived Fission Products for Waste Disposal", BNL-8558, (1964).

/37/ H.C. Claiborne, "Neutron-Induced Transmutation of High-Level Radioactive Waste", ORNL-TM-3964, (1972).

/38/ R.G. Reichert. "Transportation of Radioactive Waste-Materials into the Sun", in proceeding of 23rd Congress of the International Astronautical Federation, Vienna, Austria, 9-14 Oct. 1972.

/39/ A.M. Platt and R.W. Ramsey, "Long-Term Waste Management Methods" in proceedings of the conference Management of Radioactive Wastes from Fuel Reprocessing, Paris, Dec. 1972.

/40/ K.J. Schneider and J.H. Jarrett, "Alternative Means of Ultimate High-Level Waste Management", Trans. Am. Nucl. Soc. 17, 325-326 (1973).

/41/ I.L. Branch and J.A. Connor, "Nuclear Safety in Space", Nucleonics 19, 64-68 (1961).

/42/ E.P. Hardy et al., "Global Inventory and Distribution of Fallout Plutonium", Nature 241, 444-445 (1973).

/43/ L.E. Van Bibber and W.G. Parker, "Post-Impact Analysis of Nuclear-Waste Space Disposal Containers", Trans. Am. Nucl. Socl, 17, 328 (1973).

/44/ F.L. Culler, J.O. Blomeke and W.G. Belter, "Current Developments in Long-Term Radioactive Waste Management", in proceedings of the 4th Internation Conf. on Peaceful Uses of Atomic Energy, Geneva, Sept. 1971.

/45/ K. Kühn, W. Diefenbacher, H. Krause, H. Schmidt, "Disposal of Solidified High-Level Radioactive Wastes in the Asse Salt Mine", in proceedings of the conference Management of Radioactive Wastes from Fuel Reprocessing, Paris, Dec. 1972.

/46/ H. Krause, R. Randl, "Treatment and Final Disposal of Radioactive Wastes from Fuel Reprocessing in the Federal Republic of Gernamy - A Survey on Policy and R + D Work", in proceedings of the conference Management of Radioactive Wastes from Fuel Reprocessing, Paris, Dec. 1972.

/47/ M.J. Bell and R.S. Dillon, "The Long-Term Hazard of Radioactive Wastes Produced by the Enriched Uranium, Pu - ^{238}U, and ^{233}U - Th Fuel Cycles", ORNL-TM-3348, (1971).

/48/ "Gabun: Urzeitliche natürliche Kellenreaktion", Atomwirtschaft, 17, p. 545, Nov. 1972.

/49/ "Le 'Réacteur Fossile' d'Oklo (Gabon)", Energie Nucleaire 14 376-381 (1972).

/50/ "Zur Sicherheit der Endlagerung radioaktiver Abfälle im Salzbergwerk Asse", Gesellschaft für Strahlen und Umweltforschung mbH, München (1973).

/51/ K.J. Schneider, "Solidification and Disposal of High-Level Radioactive Wastes in the United States", Reactor Technology 13, 387-415 (1971).

/52/ U.S. Atomic Energy Commission, "The Safety of Nuclear Power Reactors and Related Facilities," WASH-1250, July 1973.

/53/ Barnwell Nuclear Fuel Plant - Environmental Report, DOCKET NUMBER 50-332-27, Allied-Gulf Nuclear Services, Nov. 1972.

/54/ F. Gera and D.G. Jacobs, "Considerations in the Long-Term Management of High-Level Radioactive Wastes", ORNL-4762 (1972).

/55/ V.E. Blake, "A Prediction of the Hazards from the Random Impact of Meteorites on the Earth's Surface", SC-RR-68-838, (1968).

/56/ J.J. Cohen, A.E. Lewis and R.L. Braun, "In Situ Incorporation of Nuclear Waste in Deep Molten Silicate Rock", Nuclear Technology 13, 76-88 (1972).

/57/ S.J. Rimshaw and E.E. Ketchen, "Curium Data Sheets", ORNL-4357, Jan. 1969.

/58/ E.D. Clayton, "Fissionability and Criticality: From Protactinium to Californium and Beyond", Nucl. Sci. Eng. 52, 417-420 (1973).

/59/ E.J. Zeller and D.F. Saunders, "A Suggestion for a Permanent International Polar High-Level Radioactive Waste Repository", COO-1057-59 (1972).

/60/ "Radioactive Wastes on Ice; Further Discussion", Bulletin of the Atomic Scientists, April 1973.

/61/ E.J. Zeller, D.F. Saunders, and E.E. Angino, "Putting Radioactive Wastes on Ice", Bulletin of the Atomic Scientists, Jan. 1973.

/62/ J. Donea, S. Giuliani and L. Kestemont, "Processus d'auto-enfouissement de recipients contenant des residus radioactifs de haute activité", EUR-4903 f (1972).

/63/ S.E. Logan, "Deep Self-Burial of Radioactive Wastes by Rock Melting Capsules", Tran. Am. Nucl. Soc., 16 177-178 (1973).

/64/ M.J. Szulinski, J. H. Warren and O.J. Elgert, "Engineered Storage of Radioactive Wastes", Proc. of the Conference Management of Radioactive Wastes from Fuel Reprocessing, Paris, Dec. 1972.

/65/ J.O. Blomeke and L.C. Emerson, "A Safety Analysis of Tank Storage", in Waste Treatment and Disposal Semiannual Progress Report, July-December 1965, ORNL-TM-1465 (1966).

/66/ ORNL Staff, "Siting of Fuel Reprocessing Plants and Waste Management Facilities", ORNL-4451, (1970).

/67/ Wm L. Lennemann, "Management of Radioactive Aqueous Wastes From U.S. Atomic Energy Commission's Fuel Reprocessing Operation, Experience and Planning", Proc. of the Conference Management of Radioactive Wastes from Fuel Reprocessing, Paris, Dec. 1972.

/68/ "The Great Escape at Hanford", Nuclear News, 16, p. 82, Sept. 1973.

/69/ L.D. Maxim and C.L. Brazie, "A Multi-Stage Input-Output Model for Evaluation of the Environmental Impact of Energy Systems", IEEE Transactions on Systems, Man and Cybernetics, (1973).

/70/ J.B. Rivard, "Risk Minimization by Optimum Allocation of Resources Available for Risk Reduction", Nuclear Safety, 12, 305-309 (1971).

Abbreviations

LWR : light water thermal reactors consisting of 50% pressurized water and 50% boiling water reactors

PWR : pressurized light water reactor

BWR : boiling light water reactor

LMFBR: liquid metal (sodium) cooled fast breeder reactor with oxide plutonium fuel

HTGR : high temperature gas cooled thermal reactor operating with a 93% ^{235}U fuel

THTR : high temperature gas cooled thermal reactor operating with ^{233}U recycle fuel and ^{235}U makeup

BRD : Bundesrepublik Deutschland (Federal Republic of Germany)

Ci : curie (3.7×10^{10} disintegrations/second)

KCi : kilocurie (curie $\times 10^{3}$)

MCi : megacurie (curie $\times 10^{6}$)

MWD/T: burnup designation in megawatt days per ton (1000 kg) of fuel

kw : kilowatts (watts $\times 10^{3}$)

Mw : megawatts (watts $\times 10^{6}$)

GW : gegawattes (watts $\times 10^{9}$)

kg : kilograms (grams $\times 10^{3}$)

Pa : element symbol for protactinium

U : element symbol for uranium

Np : element symbol for neptunium

Pu : element symbol for plutonium

Am : element symbol for americium

Cm : element symbol for curium

DF : decontamination factor

HI : hazard index of radioactive wastes

Q_i : symbol for amount of curies of a particular quantity of radioactive isotope i

MPC_x^i : maximum permissible concentration of isotope i in water (x=w) or air (x=a) for general population

MPI_x^i : maximum permissible ingestion (x=w) or inhalation (x=a) of isotope i for the general population, per year

T^i : physical half-life of isotope i

λ^i : $0.693/T^i$

P_i : probability of isotope i leaving a waste disposal site and reaching man

a^i, b^i : weighting factors used in the definition of a hazard index (HI) for radioactive wastes

F.P. : fission products

cm : centimeter

m : meter

km : kilometer ($m \times 10^3$)

MT : metric ton (1000 kg)

ICRP : International Commission on Radiological Protection

WU : waste unit, the quantity of high-level wastes from the reprocessing of one ton of spent year

m_x^3 : cubic meters of water (x=w) or air (x=a)