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RADIOLOGICAL IMPACT OF RADIOACTIVE WASTE MANAGEMENT**

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1. INTRODUCTION

1. The management of the radioactive wastes from the nuclear fuel cycle must take into account the requirements of radiation protection. Some of these requirements are concerned with the protection of individuals, achieved by the limitation of individual radiation exposures. Other requirements relate to the total radiation impact incurred from the practice under consideration, and involve consideration of collective radiation exposures.

2. It is not the purpose of this document to prove that the management of radioactive wastes can be carried out within the requirements of individual exposure limitation and under sufficiently safe conditions. These facts are the subject of a very extensive literature and several national and international reviews. This document presents some assessments of the collective radiation impact which could be expected from the adequate management of the radioactive wastes from the nuclear fuel cycle. In order to provide perspective, also the maximum individual exposures are estimated for the major contributions, and when relevant, parts of the collective impact (integrated over the near future only), are also presented.

3. The assessments presented in this document are not site-specific, but they are attempts to obtain representative values of the collective dose commitments resulting from disposal of radioactive wastes, in order to allow comparisons between different fuel cycle alternatives and with the commitments resulting from other operations of the cycle, which are better known and already placed

in the proper perspective provided by other human exposures to radiation (1). Comparative assessments over the entire fuel cycle, however, would require an in-depth evaluation of all radiation exposures, including occupational exposures and exposures due to radioactive effluents, which are only treated superficially in this document, in so far as they relate to waste management operations.

4. Comparisons between collective dose commitments from different parts of the nuclear fuel cycle are facilitated if the values are normalized. As the scale of all parts of the cycle is related to the electrical energy generation they serve, it seems reasonable to use this quantity to normalize the collective dose commitment from waste disposal, as it is customarily done for the other operations of the cycle (1). The assessments in this document will therefore be expressed in terms of collective dose commitments per unit of electrical energy generated (GW(e)y).

5. As indicated by the heading, the quantitative evaluation of the health and safety impact is limited to the radiation impact originating from the radioactive wastes whether processed, stored or disposed. An assessment of other components of a comprehensive safety evaluation would include industrial accidents in the construction and operation of the installations, in the transportation and in the excavation and operation of the repositories. A detailed assessment of this type was not considered possible within the time and resources available. For a given level of technology it is obvious that fuel cycle strategies requiring the handling of larger amounts of mill tailings, the mining of additional repository space or the physical handling of larger

waste quantities would probably result in more industrial type accidents*.

2. CONCEPTS USED FOR QUANTIFYING RADIOLOGICAL IMPACTS

6. This document uses the concept of the weighted mean whole body exposure, introduced by the ICRP in its publication 26 (2). This concept is based on the principle that at a given level of safety the risk should be equal whether the whole body is irradiated uniformly or whether there is nonuniform or partial irradiation. This condition is met by applying the radiation protection requirements to the effective dose equivalent, H_E,

$$H_E = \sum_T w_T H_T$$

where w_T is a weighting factor representing the fraction of risk resulting from tissue T when the whole body is irradiated uniformly, and H_T is the dose equivalent in tissue T. The values of w_T recommended by the

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* In a background document (76) for the INFCE WG 7 a rough estimate of industrial accident incidence is derived for the construction and operation over a 40 year period of a salt repository handling the waste from a 100 GWa nuclear power system. According to this reference the estimated total number of fatalities would vary between 7-16 while the corresponding numbers of disabling injuries would be 350-700.

ICRP are:

Gonads	0.25
Breast	0.15
Red bone marrow	0.12
Lung	0.12
Thyroid	0.03
Bone	0.03
Remainder	0.30

Regarding the remainder, the ICRP recommends that a value of $w_T = 0.06$ is applied to each of the five organs receiving the highest dose equivalent, and that the exposure of the other remaining organs is neglected. The gastrointestinal tract is treated as four separate organs (stomach, small intestine, upper large intestine and lower large intestine). The skin, lens of the eye, hands, forearms, feet and ankles are not included in the remainder.

7. Some additional quantities (3, 4) are necessary for the discussions presented in this section regarding the exposure of populations. The exposure of a population in many cases is delivered at a varying rate over a period of time. In these cases it is convenient to define a quantity which represents the distribution of dose rate over the population, being equal to the product of the number of exposed individuals and the average dose rate. The collective effective dose equivalent rate, $\dot{S}_E(t)$, is defined as

$$\dot{S}_E(t) = \int_0^\infty \dot{H}_E(t)\, N(\dot{H}_E, t)\, d\dot{H}_E$$

where $N(\dot{H}_E, t)$ is the population spectrum in effective dose equivalent rate at time t, $N(\dot{H}_E, t)\, d\dot{H}_E$ being the number of individuals receiving at time t an effective dose equivalent rate in the range $\dot{H}_E(t)$ to $\dot{H}_E(t) + d\dot{H}_E$.

The total collective dose equivalent rate from a source k, $\dot{S}_{E,k}$, is obtained by including in the population under consideration all individuals receiving a dose from the source k.

8. In some cases a population becomes committed to receive radiation exposures by the introduction of a practice, the adoption of a decision or some other finite originating event. The exposure of the population could be delivered over considerable time after the originating event. In order to have a measure of the total exposure, of the population, the collective effective dose equivalent commitment is used. The collective effective dose equivalent commitment, $S^c_{E,k}$, due to a given event k is defined as the infinite time-integral of the collective effective dose equivalent rate, $\dot{S}_{E,k}(t)$, caused by that event

$$S^c_{E,k} = \int_o^\infty \dot{S}_{E,k}\,(t)\,dt.$$

In this section of the report, the terms collective dose commitment, unless specifically qualified, are taken to mean the collective effective dose equivalent commitment. Similarly, the terms normalized collective dose commitment are used for the quantity collective effective dose equivalent commitment per unit electrical energy generated.

9. The assessment of the collective effective dose equivalent commitment from an originating event is particularly useful for two purposes. On one hand it can be used in relative detriment assessments, on the assumption that the risk of radiation induced deleterious effects is proportional to the effective dose equivalent. On the other hand, it can also be used to assess future exposures from continued practices, which could be considered as sequences of events each delivering exposures over times

exceeding the duration of the event. Both uses have to be qualified in the case of exposures delivered over very long times, as is the case with waste disposal.

10. The first use, relative detriment assessment, is the main application of the quantity in this section of the report. It is used to quantify the radiological impact from waste management and to compare it with those from other operations of the fuel cycle. It implies, however, a comparative judgement of the importance of deleterious effects at present or the near future, and those which conceivably might be induced in the very far future, after periods much longer than the history of man. It gives the same weight to present and far-future detriments, which is not the usual practice in other types of human judgements. Furthermore, as little is really known about the environmental parameters which will govern the far-future exposures, considerable uncertainty is attached to the assessments.

11. The second use, assessment of future maximum per caput dose rates, is based on the fact that in the case of a continued practice the resulting average (per caput) dose rate will increase and eventually reach a steady state. However, in the case of waste disposal, as in other exposures delivered over very long time, it would not be realistic to assume a continued practice for such long time as required by the per caput dose rate to approach steady state. In these cases, the maximum per caput dose rate which will be experienced in the future can be assessed by procedures which have been discussed in the literature (3, 4, 5, 6). It can be shown that, in these cases, the maximum per caput dose rate to be experienced in the future is approximated by

$$\dot{H}_{max} \simeq \frac{R}{N} \int_o^{\tau} \dot{S}_{E,1}(t)\, dt$$

where R is the practice rate, N the population size, τ is the duration of the practice, and $\dot{S}_{E,1}(t)$ is the collective dose rate per unit practice, as a function of time. The usual name of the above integral is the incomplete collective dose commitment. For assessments related to the production of power by fission, a value of 500 years has been used in most recent assessments (1). In the case of very long-lived nuclides which disperse about homogeneously in the environment around the globe, the above expression can be approximated by

$$\dot{H}_{max} \simeq \frac{R}{N} \int_{t_d}^{\tau} \hat{\dot{S}}_{E,1}\, e^{-\frac{t}{T}}\, dt$$

where $\hat{\dot{S}}_{E,1}$ is the collective dose rate at the time t_d when the dispersion is completed, and T is the mean life of the nuclide. Provided $\tau > t_d$ and $\tau << T$,, the equation can be approximated linearly by

$$\dot{H}_{max} \simeq \frac{R}{N}\, \tau\, \frac{S^c_{E,1}}{T}$$

showing that the maximum per caput dose rate from these long-lived nuclides depends on the total energy which can be extracted from the fuel resources ($R\,\tau$) and not on the actual rate of its generation.

3. RADIOLOGICAL IMPACT FROM MANAGEMENT OF MILL WASTES

12. The fuel cycle strategies discussed in this report require an input of uranium and, in some cases, also of

thorium. From the radionuclides contained in the residues of thorium ore processing, it follows that the radiological contribution of the thorium mill tailings can be neglected compared to those of the uranium mill tailings, in the case of the fuel cycles using thorium. The present chapter, therefore, deals only with the radiological impact of uranium mill wastes.

Uranium processing mills are usually located near the mining areas to avoid transportation of large quantities of what is ultimately waste materials. In some cases the ore is locally treated by heap-leaching and precipitated pre-concentrates are then transported to the mills. Releases of radioactive materials during milling operations occur as gaseous and liquid effluents, but the bulk of the activity originally present in the ore is undisolved and is stored in the solid mill tailings.

13. The radionuclides contained in the ore include uranium nuclides and uranium decay products, such as ^{230}Th and ^{226}Ra. For ore containing 2 kg U per tonne, it can be readily calculated, assuming secular equilibrium, that one tonne of ore contains an activity of $6.7\ 10^{-4}$ Ci from each member of the ^{238}U chain, adding to a total activity of $9.3\ 10^{-3}$ Ci.

14. In the mills, the ore is blended, crushed, wet ground and then subjected to acid or alkaline leaching procedures, according to the ore characteristics. The soluble uranium is recovered by solvent extraction or by precipitation. About 14% of the total activity contained in the ore fed to the mill accompanies the uranium concentrated (7), uranium itself being recovered in about 95%. The uranium concentrate contains also about 0.2% of the radium and 5% of the total thorium present originally in

the ore (8). Unsupported short-lived daughters, such as most of ^{234}Th (^{234m}Pa), disappear from the mills waste by decay, due to the removal of the long-lived parent nuclides. The remaining, about 70% of the total activity originally present in the ore, is largely undissolved and is contained in the solid mill tailings, the activity being about $6\ 10^{-3}$ Ci per tonne of tailings, allowing for the residues of extraction agents.

15. The activity of the nuclides present in the solid tailings of the mill can be evaluated for the different fuel cycle strategies (9), using the information discussed in paragraphs 12 and 13. Definition of the strategies and the results of this assessment normalized per GW(e)y generated in each strategy are presented in Table 1.

16. The tailings area remaining after the mill becomes inactive is a long term source of environmental releases of radionuclides, due to radon emanation and erosion by wind and water. Emanation from the tailings is primarily from radon produced in the surface layer, about 90% originating in the top 2 m (10, 11).

17. The tailings area can be estimated from past practice as about 0.1 m^2 per tonne of tailings (11), corresponding to $1.5\ 10^4$, $8.5\ 10^3$, $8.6\ 10^1$, $1.3\ 10^4$, $5.3\ 10^3$, $5.0\ 10^2$ and $4.0\ 10^3$ m^2 per GW(e)y respectively for the alternative fuel cycle strategies listed in Table 1. The input of radon into the atmosphere from the tailings can be assessed using a nominal release factor of about $5\ 10^{-10}$ Ci m^{-2} s^{-1}, for uncovered tailings (11). It should be recognized, however, that some differences in emanation can be observed between arid and humid areas. Covering the tailings pile with soil reduces the radon emanation rate by a factor of two for each metre of cover.

Table 1.

Activity of radionuclides in uranium mill tailings
normalized per unit electrical energy

Nuclide	Fuel Cycle Strategy						
	(1)	(2)	(3)	(4)	(5)	(6)	(7)
	Normalized activity Ci per GW(e)y						
^{238}U	$3.6\ 10^{0}$	$2.1\ 10^{0}$	$2.1\ 10^{-2}$	$3.1\ 10^{0}$	$1.3\ 10^{0}$	$1.2\ 10^{-1}$	$1.0\ 10^{0}$
^{234}Th	$3.6\ 10^{0}$	$2.1\ 10^{0}$	$2.1\ 10^{-2}$	$3.1\ 10^{0}$	$1.3\ 10^{0}$	$1.2\ 10^{-1}$	$1.0\ 10^{0}$
^{234m}Pa	$3.6\ 10^{0}$	$2.1\ 10^{0}$	$2.1\ 10^{-2}$	$3.1\ 10^{0}$	$1.3\ 10^{0}$	$1.2\ 10^{-1}$	$1.0\ 10^{0}$
^{234}U	$3.6\ 10^{0}$	$2.1\ 10^{0}$	$2.1\ 10^{-2}$	$3.1\ 10^{0}$	$1.3\ 10^{0}$	$1.2\ 10^{-1}$	$1.0\ 10^{0}$
^{230}Th	$6.8\ 10^{1}$	$4.0\ 10^{1}$	$4.0\ 10^{-1}$	$6.0\ 10^{1}$	$2.5\ 10^{1}$	$2.4\ 10^{0}$	$1.9\ 10^{1}$
^{226}Ra	$7.2\ 10^{1}$	$4.2\ 10^{1}$	$4.2\ 10^{-1}$	$6.3\ 10^{1}$	$2.6\ 10^{1}$	$2.6\ 10^{0}$	$2.0\ 10^{1}$
^{222}Rn	$7.2\ 10^{1}$	$4.2\ 10^{1}$	$4.2\ 10^{-1}$	$6.3\ 10^{1}$	$2.6\ 10^{1}$	$2.6\ 10^{0}$	$2.0\ 10^{1}$
^{218}Po	$7.2\ 10^{1}$	$4.2\ 10^{1}$	$4.2\ 10^{-1}$	$6.3\ 10^{1}$	$2.6\ 10^{1}$	$2.6\ 10^{0}$	$2.0\ 10^{1}$
^{214}Pb	$7.2\ 10^{1}$	$4.2\ 10^{1}$	$4.2\ 10^{-1}$	$6.3\ 10^{1}$	$2.6\ 10^{1}$	$2.6\ 10^{0}$	$2.0\ 10^{1}$
^{214}Bi	$7.2\ 10^{1}$	$4.2\ 10^{1}$	$4.2\ 10^{-1}$	$6.3\ 10^{1}$	$2.6\ 10^{1}$	$2.6\ 10^{0}$	$2.0\ 10^{1}$
^{214}Po	$7.2\ 10^{1}$	$4.2\ 10^{1}$	$4.2\ 10^{-1}$	$6.3\ 10^{1}$	$2.6\ 10^{1}$	$2.6\ 10^{0}$	$2.0\ 10^{1}$
^{210}Pb	$7.2\ 10^{1}$	$4.2\ 10^{1}$	$4.2\ 10^{-1}$	$6.3\ 10^{1}$	$2.6\ 10^{1}$	$2.6\ 10^{0}$	$2.0\ 10^{1}$
^{210}Bi	$7.2\ 10^{1}$	$4.2\ 10^{1}$	$4.2\ 10^{-1}$	$6.3\ 10^{1}$	$2.6\ 10^{1}$	$2.6\ 10^{0}$	$2.0\ 10^{1}$
^{210}Po	$7.2\ 10^{1}$	$4.2\ 10^{1}$	$4.2\ 10^{-1}$	$6.3\ 10^{1}$	$2.6\ 10^{1}$	$2.6\ 10^{0}$	$2.0\ 10^{1}$

NOTE: (1) Light Water Reactor with spent fuel disposal
(2) Light Water Reactor with plutonium recycle
(3) Fast Breeder Reactor with plutonium recycle
(4) Heavy Water Reactor with spent fuel disposal
(5) Heavy Water Reactor with plutonium recycle
(6) Heavy Water Reactor with uranium-thorium recycle
(7) High Temperature Reactor with uranium-thorium recycle

18. The assessment of the contribution made by the radon released from the tailings to the normalized collective dose commitment entails some guesswork. Radon emanation could conceivably continue for several hundred thousand years, as the activity of ^{226}Ra is in equilibrium with that of ^{230}Th. A small part of the emanation could even continue for millions of years because of the remaining ^{238}U precursor. However, the continued presence of the tailings pile on the surface with the presence of the parent nuclides, for both such periods seems very unlikely.

19. For the present assessment it will be assumed as an extreme case that the tailings pile remains undisturbed for times comparable with the mean life of ^{230}Th, and that the tailings are covered with 2 m of earth, practice which seems likely to be adopted for future operations. Regarding population distribution around mills, past experience shows that it is generally sparse and that little growth can be expected. Values of 3 per km^2 at short distances (up to about 80 km) and 25 per km^2 at a greater distance have been taken as representative (1).

20. An estimate of the collective dose commitment due to radon release from the tailings of an inactive mill can be obtained by comparison with the natural radon emanation from the soil and the corresponding radiation doses. It can be shown (1) that for comparisons of a limited emanating area (the tailings) with a practically infinite natural emanating area, the following relation applies:

$$S_E^c = \dot{H}_{E,n} \frac{Q_o}{A_n} N_\delta \tau$$

where S_E^c is the collective dose commitment, $\dot{H}_{E,n}$ is the effective dose equivalent rate due to the natural emana-

tion of radon, Q_o is the activity emanation rate from the tailings, A_n is the natural activity emanation rate per unit area of soil, N_δ is the population density and τ is the mean emanating time of the tailings.

21. The assumed earth cover would reduce emanation from the tailings by a factor of four. Using the tailings areas and release factor given in paragraph 17, together with representative natural values for emanation and dose rate (1) and the weighting factor $w_T = 0.12$ for lung doses (paragraph 6), the following dose commitment estimates are obtained for the different fuel cycle strategies for the assumed extreme case of tailings undisturbed for periods comparable with the mean life of ^{230}Th.

Strategy	Normalized collective dose commitment (man rem/GW(e)y)
1	$1.1\ 10^5$
2	$6.5\ 10^4$
3	$6.2\ 10^2$
4	$9.7\ 10^4$
5	$4.0\ 10^4$
6	$4.0\ 10^3$
7	$3.0\ 10^4$

22. These values are clearly overestimates, because they assume tailings emanation undisturbed for the order of 10^5 years. Experience over much shorter periods would suggest that the area is likely either to be eroded and the materials transported finally into waters, or, in some cases, continually covered with further dust. In both cases emanation would be substantially reduced. Assuming an effective mean residence time for ^{230}Th and ^{226}Ra of the order of 10^3 years, the estimated collective dose commitments mediated through radon emanation are:

Strategy	Normalized collective dose commitment (man rem/GW(e)y)
1	$9.6\ 10^{2}$
2	$5.6\ 10^{2}$
3	$5.3\ 10^{0}$
4	$8.4\ 10^{2}$
5	$3.5\ 10^{2}$
6	$3.4\ 10^{1}$
7	$2.6\ 10^{2}$

23. If ^{230}Th and ^{226}Ra are assumed to move away from the tailings, as it was shown in paragraph 22, it is interesting to estimate the additional collective dose commitment contribution expected from the entry of these nuclides into circulating waters. The models used for this estimation are discussed in detail in section 4.3.1. Under the assumption that all the ^{230}Th and ^{226}Ra are involved in this process, the following collective dose commitment contributions are calculated as upper estimates for the water pathways:

Strategy	Normalized collective dose commitment (man rem/GW(e)y)
1	$3.6\ 10^{4}$
2	$2.1\ 10^{4}$
3	$2.1\ 10^{2}$
4	$3.2\ 10^{4}$
5	$1.3\ 10^{4}$
6	$1.3\ 10^{3}$
7	$1.0\ 10^{4}$

24. Radium-226 is the major contributor to the above estimates, its passage through fresh waters resulting in about 76% of the values. This fact is the cause of major uncertainty in the results for a generic assessment, because the presence of fresh water pathways will necessarily be site specific. It should be noted, regarding the combined commitment from radon emanation and radium ingestion, that only very prolonged residence of the ^{230}Th-^{226}Ra pair in the tailings site would reduce by decay the collective dose commitment due to water pathways, but at a cost of increasing the inhalation contribution, to values which would not exceed those of paragraph 21.

25. Representative values of the collective dose commitments from mill tailings can be obtained by adding the contributions from radon emanation (paragraph 22) and from the water pathways (paragraph 23). Much longer retention of nuclides in the tailings would change the relative significance of both components, but would have little influence on the totals.

Strategy	Normalized collective dose commitment (man rem/GW(e)y)
1	$3.7\ 10^4$
2	$2.2\ 10^4$
3	$2.2\ 10^2$
4	$3.3\ 10^4$
5	$1.3\ 10^4$
6	$1.3\ 10^3$
7	$1.0\ 10^4$

26. In spite of all uncertainties, the contribution of mill tailing to the collective dose commitment is important

when compared with the contribution of occupational exposure and effluents from different parts of the fuel cycle (1). The contribution, as expected, depends on the uranium requirements for each reference fuel cycle, being the largest for once-through cycles, and smaller by a factor of 2 for the corresponding recycling strategies, by two orders of magnitude for the fast breeder fuel cycle, and one order of magnitude for the heavy water thorium-uranium cycle. On the other hand, only a smaller reduction factor is provided by the high temperature thorium-uranium cycle. Compared with the exposure of the world population to natural radiation sources, the contributions per GW(e)y are equivalent to about 1 hour or less of natural exposure.

27. As it was discussed in section 2, the incomplete collective dose commitment is sometimes calculated as a tool for assessing the maximum per caput dose rate in the future. The incomplete commitment is integrated over the time of practice, usually assumed to cover a period of the order of 100 years for the specific case of uranium mining and milling (1). The incomplete commitment contributions from inhalation pathway would be one order of magnitude smaller than the values shown in paragraph 22. Thorium-230 might remain in the tailing over that time. However, assuming again a mean effective residence time for ^{230}Th-^{226}Ra of about 10^3 years, the water pathways would deliver a collective dose rate which integrated over the 100 years of highest values, results also in one order of magnitude less than the values attributed to ^{226}Ra through fresh waters (paragraph 23). The incomplete collective dose commitments from mill tailing can therefore be estimated as comprised within the following ranges:

Strategy	Normalized incomplete collective dose commitment (man rem/GW(e)y)	
1	$9.6\ 10^{1}$	$2.8\ 10^{3}$
2	$5.6\ 10^{1}$	$1.6\ 10^{3}$
3	$5.3\ 10^{-1}$	$1.6\ 10^{1}$
4	$8.4\ 10^{1}$	$2.4\ 10^{3}$
5	$3.5\ 10^{1}$	$1.0\ 10^{3}$
6	$3.4\ 10^{0}$	$1.0\ 10^{2}$
7	$2.6\ 10^{1}$	$7.7\ 10^{2}$

28. The individual doses to be experienced from mill tailings in the future are very dependent of local conditions. Based on published generic assessments (1), and using the techniques outlined in section 2, a value of a few tens of microrem per year is considered to be representative of the maximum per caput doses in the future for members of the public.

4. RADIOLOGICAL IMPACT FROM MANAGEMENT OF WASTES FROM FUEL CYCLE STEPS FOLLOWING REACTOR OPERATIONS

29. The bulk of the radioactivity associated with the production of energy by nuclear fission is contained in the spent fuel and is considered waste except for nuclides of uranium and plutonium in the strategies involving reprocessing. This waste must be subjected to suitable treatment, some retrievable storage and eventual disposal in which control over the waste is relinquished. Disposal is necessarily involved due to the very long half lifes of some of the relevant nuclides, much longer than any conceivable administrative control which could be established by the present society. Storage and dispo-

sal problems still exist even if the spent fuel is not reprocessed, the wastes being in this case merely in a different form. In addition to these wastes, reactor operations generate other wastes, including discarded components, which represent a problem because of their volume but which have a radioactivity of small relative importance compared to that of the spent fuel or of the high level wastes from fuel reprocessing.

30. Disposal strategies for the spent fuel and, mainly, for the high level liquid wastes have been discussed in detail (12, 13, 14, 15, 16, 17, 18), being agreed that the high level liquid wastes should eventually be solidified prior to disposal, this being the reference solution adopted.

4.1. Isolation Concept

31. The basic radiological safety concept underlying disposal in deep geological formations is the isolation of the high-level wastes from man's environment until the potential radiation exposures are reduced to levels compatible with the requirements of radiological protection. This required delay is provided by several engineered features and by the time of migration of the relevant nuclides until they reach surface waters. Main engineered delays are represented by the integrity of the waste containers and the stability of the waste matrix, either the spent fuel or solids in the form of suitable glass. Container life, leachability of matrix and nuclide migration are considered in this section only in relation with the order of magnitude of the delay time they assure between disposal and input of the waste radioactive materials into man's environment.

4.1.1. Engineered Aspects of Isolation

32. Different materials can be considered for containers, reflecting different degrees of conservatism in the proposals. Stainless steel alone would make the corrosion by borosilicate glass very small, specially as the temperature falls below several hundred degrees centigrads (19), and this solution has been selected for the assessments. It should be noted that, if found necessary, several technical possibilities are available for more conservative containers (5). There are materials such as titanium and lead which are very resistant to attack by groundwaters or sea-water. Containers constructed with an external layer of titanium, a central large layer of lead and an internal lining of stainless steel are, therefore, expected to have a substantial long life, exceeding a few thousand years. For the purpose of the collective dose assessment presented in this document, it is assumed that the mean life of containers is over 1000 years (5, 38), sufficient for the decay of most fission products.

33. The solidified wastes are assumed to be in the form of borosilicate glass. A large amount of information on the characteristics and behaviour of the glass has been reported (19, 20, 21, 22, 23, 24, 25, 26, 27, 28, 29, 30, 31, 32, 33, 34, 35, 36, 37). The thermal history of waste appears to have no effect on the leach rate, nor on the internal distribution of the contained activity. Leach rates seem to be the same in destilled water and in sea-waters, and it seems unlikely that they would be different in groundwaters. At 25°C, measured leach rates are in the range 10^{-9} to 10^{-4} g/cm^2 day, and a dependency on the temperature of the leaching solution has been shown, the rate increasing by a factor of about 30 as the temperature approaches 90°C. On the other hand, the effect of the irradiation of the glass and of the energy released by decay seems to have a relatively small effect on leach rate. For the purposes of the assessments presented in

this document a leach rate of about $4\ 10^{-4}$ g/cm^2 year is assumed to apply during the entire life of the disposed waste. However, in view of the very low water flow rate (section 4.1.2), this assumed leach rate, which is larger than the mean in the quoted range, may be an overestimate leading to conservative assessments.

34. Thermal stresses may cause fracture of the glass during cooling. In addition alpha-decay, which produces a build-up of helium in the glass, has been indicated as a possible cause of glass fractures. As the net amount of glass material leached per unit time depends on the surface area of the glass blocks, some assumptions are necessary on their size. As in other published assessments (38), it has been assumed that the glass fractures into pieces of about 10 cm diameter. In these conditions, complete dissolution of the glass, assuming the presence of ground-water, would take over $3\ 10^4$ years, and the mean age of the released nuclide mixture would be about $8\ 10^3$ years. Container and glass can, therefore, be assumed to provide together a delay at least of the order of 10^4 years, which would operate over the entire mixture of waste radionuclides, not disturbing the variation of nuclide composition due to decay and parent-daughter relationships as a function of time. As indicated in the previous paragraph, the assessed delay may be a conservative estimate, the actual values being in the range 10^4- 10^5 years. This fact is reinforced by the use of back filling in the repository.

35. In the case of direct disposal, the spent fuel is placed directly in final disposal without prior reprocessing. The leaching characteristics in this case have not been studied in detail, but UO_2 pellets have low rates of release of contained nuclides when in contact with water (12). There have been technical studies for adequate canister materials, and as the inventory of radionuclides is larger than in separated high level

wastes, the intended life of the canisters is longer. Steel canisters have been adopted as the reference solution. More conservative solutions would be available if found necessary. Copper canisters (12) have been particularly studied, and with a thickness of 20 cm, the penetration of the canister has been estimated to take about 10^5 years, although some conservative assessments give a shorter time of less than about 10^4 years. Aluminium oxide ceramic is also being studied for spent fuel canisters. For the assessments presented in this document, it will be assumed that the engineered isolation provided by canister and spent fuel matrix is as good as that described in previous paragraphs, the delay time being of the order of 10^4 to 10^5 years.

4.1.2. Geological Aspects of Isolation

36. Since the wastes contain extremely long-lived nuclides, it can be assumed that a small portion of the initial radioactivity will outlive the container and the waste matrix. Transport by circulating ground waters is the most realistic mechanism by which waste nuclides could return to man's environment. Ground-water flow follows Darcy's law and depends on permeability and hydraulic gradient.

37. For a repository in a well selected site in unfractured cristalline rock (5, 39) the resulting groundwater flow should be very small and the permeability would not be measurable by conventional means. Indirect evaluations could be obtained by geochronological assessments of the age of the groundwater (5). Assuming flow rates of a few tens of cubic centimetres per square meter and year, the travel time of water across the rock formation and buffer zone should be of the order of many thousand years. As an order of magnitude, it has been

assumed that the travel time of the groundwater from the disposal site until it reaches surface waters is 10^4 years.

38. In the case of repositories in salt formations, a permanent low flow rate of groundwater is not assumed to occur from the inception (39). However, combinations of conditions leading to disruption of the repository and release of material by leaching to groundwater, are considered to be more probable than other disruptive events discussed in paragraphs 90 to 97 (40).

39. Several studies have estimated the probability of primary disruptive events associated with groundwater, for different time periods of a repository in salt formations (40). This probability is substantial for periods of the order of 10^3 - 10^4 years and becomes close to one for periods of the order of 10^5 - 10^6 years. It should be noted that such events associated with groundwaters do not lead to instantaneous release of nuclides, but are instead the start of a slow leaching process into circulating groundwaters.

40. Once leached into groundwaters, the migration of radionuclides in most cases should be considerably slower than that of water. If exchange phenomena operate, the delay factor relating the travel time of a nuclide and that of water would be equal to $1 + K_d \delta/p$, where K_d is the distribution constant, δ is the density of groundwater and p is the effective porosity of the travel medium (6,12). Very little is known about distribution factors in the relevant cristalline rock media, but from extrapolation from known values in other media it could be assumed that the delay factor would be around 1 for iodine, over 100 for lantanides and probably actinides, and even larger for caesium (12).

41. Nuclide migration, however, is even more complicated than outlined above. Plutonium and several actinides can form colloids, and migration in this case would be dominated by filtration and surface adsorption. Controlling factors would be colloid particle size and dimensions of interstices through which water circulates (18). Furthermore, migration characteristics would change according to the physical and chemical properties of the medium as migration proceeds. The formation of daughter nuclides of different characteristics during migration compounds the difficulty for assessment. The solubility of uranium and radium is proportional to water temperature and also varies with pH, redox potential and the presence of CO_2 or bicarbonate ions.

42. In view of the uncertainties summarized in the previous paragraphs, and the possibilities of change of the medium characteristics in the very long time involved, it is not possible in a generic assessment to make a realistic estimate of the delay involved in the migration of different groups of waste nuclides. While ^{129}I and ^{99}Tc will probably travel at the same speed as ground water, most of the other relevant nuclides will be delayed in relation to the water. The delay factor would range between a few tens to some hundreds, and the mixture entering surface waters cannot be described by a single age. As no realistic values of the delay provided by geological isolation for each waste nuclide are available, the assessments presented in this document will not attempt to use sophisticated compartment analysis of the time functions of the resulting biospheric contamination. That type of analysis would be more applicable to site-specific assessments (5).

43. The collective dose commitment, however, can be roughly estimated to lie between two extreme values, corresponding to the entry into circulating waters of

an undisturbed mixture of waste nuclides of an age taken to be equal to either the shortest delay or the longest delay found for nuclides of the mixture. Taking into account the engineered isolation factors and the time of water migration it will be assumed that 10^5 years is representative of the delay without the effect of sorption of radionuclides, while a delay of the order of 10^6 years will be taken to represent the case where sorption is the main retardation factor. In the assessments of collective dose commitments due to waste repositories, it will be postulated that the realistic value lies in the range between the values obtained for water mixtures of 10^5 to 10^6 years reaching circulating waters.

44. For a defined site, or for a given set of defined migration parameters, it is possible to model quantitatively the migration. In an assessment carried out specially for INFCE (72), this modelling was performed for a hypotetical repository in a reference salt formation. The release assumption involves a geological event creating a fracture, which in connection with existing gradients in the aquifer system results in the flow of brine through the repository.

45. The study uses models for a three-dimensional water flow and a unidimensional nuclide transport model, taking account of radioactive decay and soil-water exchanges. By the use of these models it is possible to predict the rate of arrival of different nuclides to fresh waters, and their total input over time.

46. Figures A1 to G14 are examples of resulting arrival rates (expressed as concentration in water, which is proportional to arrival rate, for a 100 GW(e)y inicial repository inventory). The following table summarizes

the arrival times (time for maximum rate of arrival) for fission products and actinides, the values being similar for all the fuel strategies (72):

Estimated Arrival Time of Actinides into Circulating Waters*

Nuclide	Time (y)
^{99}Tc	$2.2\ 10^{6}$
^{129}I	$6.7\ 10^{5}$
^{135}Cs	$1.2\ 10^{6}$
^{226}Ra	$3.4\ 10^{6}$
^{231}Pa	$3.4\ 10^{6}$
^{229}Th	$2.5\ 10^{7}$
^{230}Th	$3.5\ 10^{6}$
^{232}Th	$8.1\ 10^{7}$
^{233}U	$2.5\ 10^{7}$
^{234}U	$3.4\ 10^{6}$
^{235}U	$3.4\ 10^{6}$
^{236}U	$3.4\ 10^{6}$
^{238}U	$3.4\ 10^{6}$
^{237}Np	$2.6\ 10^{7}$

47. It can be noted that the order of magnitude 10^{6} years, selected for the assessment procedure outlined in paragraph 43, seems to agree acceptably well with the results of this geosphere modelling. The agreement is better reflected in the collective dose commitment assessments (section 4.3.2.).

4.2. Radionuclide Composition of the Discarded Spent Fuel or the High Level Wastes

48. The radionuclide composition of the discarded spent

* For salt repository assessed by a geosphere model (72)

fuel or the high level wastes presented in Tables 2.1, 2.2, 2.3, 2.4, 2.5, 2.6 and 2.7 have been derived from information relevant to the strategies under consideration (9). These Tables present the normalized activity of relevant fission products and actinide nuclides, including under this name uranium and transuranic nuclides and relevant products of their decay chains. For the reprocessing cases it has been assumed that the high-level wastes contain over 99% of the non-volatile fission products, 0.5% of the uranium and plutonium and essentially the total amount of the other transuranic nuclides.

4.3. Collective Dose Commitment from Waste Repositories

4.3.1. Assessment Procedures

49. Transport of the long-lived radionuclides by groundwater was considered in previous paragraphs to be the normal mechanism by which waste materials return to man's environment. The hydrological cycle consists of a complicated group of paths through which the water in nature circulates and is transformed from one state to another. One of these paths (precipitation, ground infiltration and percolation to deeper zones) results in groundwater. Groundwater may later flow out as springs or may seep into streems, finally flowing to the sea. Disposal repositories will be located in formations selected for having large volume with very small amounts of migrating groundwater, but it will be assumed that eventually long-lived waste radionuclides will follow the path indicated above. They could, therefore, first cause radiation exposures through fresh-water pathways, and then through sea-water pathways.

Table 2.1

Activity composition of relevant actinides and fission products in discarded spent fuel as a function of time, normalized per unit electrical energy, for Fuel Cycle 1 Light Water Reactor with spent fuel disposal

Nuclide[a/]	Half-life (y)	Time after removal from reactor					
		10^1 y	10^2 y	10^3 y	10^4 y	10^5 y	10^6 y
		Normalized activity (Ci per GW(e)y)					
^{99}Tc	$2.1\ 10^5$	$5.0\ 10^2$	$5.0\ 10^2$	$5.0\ 10^2$	$4.8\ 10^2$	$3.6\ 10^2$	$1.8\ 10^1$
^{129}I	$1.7\ 10^7$	$1.3\ 10^0$	$1.3\ 10^0$	$1.3\ 10^0$	$1.3\ 10^0$	$1.3\ 10^0$	$1.2\ 10^0$
^{135}Cs	$3.0\ 10^6$	$8.2\ 10^0$	$8.2\ 10^0$	$8.2\ 10^0$	$8.2\ 10^0$	$8.0\ 10^0$	$6.5\ 10^0$
^{226}Ra	$1.6\ 10^3$	-	-	$8.3\ 10^{-2}$	$3.7\ 10^0$	$3.0\ 10^1$	$1.5\ 10^1$
^{234}U	$2.5\ 10^5$	$2.6\ 10^1$	$4.3\ 10^1$	$5.9\ 10^1$	$5.8\ 10^1$	$4.8\ 10^1$	$1.4\ 10^1$
^{235}U	$7.1\ 10^8$	$5.5\ 10^{-1}$	$5.5\ 10^{-1}$	$5.6\ 10^{-1}$	$6.5\ 10^{-1}$	$9.1\ 10^{-1}$	$9.3\ 10^{-1}$
^{236}U	$2.4\ 10^7$	$9.6\ 10^0$	$9.6\ 10^0$	$9.9\ 10^0$	$1.3\ 10^1$	$1.5\ 10^1$	$1.5\ 10^1$
^{238}U	$4.5\ 10^9$	$1.1\ 10^1$	$1.1\ 10^1$	$1.1\ 10^1$	$1.1\ 10^1$	$1.1\ 10^1$	$1.1\ 10^1$
^{237}Np	$2.1\ 10^6$	$1.3\ 10^1$	$1.6\ 10^1$	$3.3\ 10^1$	$3.8\ 10^1$	$3.7\ 10^1$	$2.7\ 10^1$
^{238}Pu	$8.9\ 10^1$	$9.3\ 10^4$	$4.6\ 10^4$	$5.8\ 10^1$	-	-	-
^{239}Pu	$2.4\ 10^4$	$1.1\ 10^4$	$1.1\ 10^4$	$1.1\ 10^4$	$8.3\ 10^3$	$6.3\ 10^2$	-
^{240}Pu	$6.8\ 10^3$	$1.6\ 10^4$	$1.6\ 10^4$	$1.5\ 10^4$	$6.0\ 10^3$	$6.0\ 10^{-1}$	-
^{241}Pu	$1.5\ 10^1$	$2.2\ 10^6$	$3.1\ 10^4$	$1.3\ 10^1$	$6.3\ 10^0$	$3.3\ 10^{-3}$	-
^{242}Pu	$3.8\ 10^5$	$5.0\ 10^1$	$5.0\ 10^1$	$5.0\ 10^1$	$4.9\ 10^1$	$4.2\ 10^1$	$8.1\ 10^0$
^{241}Am	$4.3\ 10^2$	$4.8\ 10^4$	$1.1\ 10^5$	$2.6\ 10^4$	$6.3\ 10^0$	$3.3\ 10^{-3}$	-
^{243}Am	$7.7\ 10^3$	$6.5\ 10^2$	$6.4\ 10^2$	$5.9\ 10^2$	$2.6\ 10^2$	$8.0\ 10^{-2}$	-

a/ Short-lived daughters in secular equilibrium with their parent-nuclides have not been included in the table

Table 2.2

Activity composition of relevant actinides and fission products in high level waste as a function of time, normalized for unit electrical energy, for Fuel Cycle 2 Light Water Reactor with plutonium recycle

Nuclide[a]	Half-life (y)	Time after removal from reactor					
		10^1 y	10^2 y	10^3 y	10^4 y	10^5 y	10^6 y
		Normalized activity (Ci per GW(e)y)					
^{99}Tc	$2.1\ 10^5$	$5.0\ 10^2$	$5.0\ 10^2$	$5.0\ 10^2$	$4.8\ 10^2$	$3.6\ 10^2$	$1.8\ 10^1$
^{129}I	$1.7\ 10^7$	$1.3\ 10^0$	$1.3\ 10^0$	$1.3\ 10^0$	$1.3\ 10^0$	$1.3\ 10^0$	$1.2\ 10^0$
^{135}Cs	$3.0\ 10^6$	$1.2\ 10^1$	$1.2\ 10^1$	$1.2\ 10^1$	$1.2\ 10^1$	$1.2\ 10^1$	$9.5\ 10^0$
^{226}Ra	$1.6\ 10^3$	-	-	$1.9\ 10^{-3}$	$1.0\ 10^{-1}$	$8.4\ 10^{-1}$	$2.6\ 10^{-1}$
^{234}U	$2.5\ 10^5$	$2.8\ 10^{-1}$	$7.4\ 10^{-1}$	$1.7\ 10^0$	$1.7\ 10^0$	$1.3\ 10^0$	$2.2\ 10^{-1}$
^{235}U	$7.1\ 10^8$	$6.0\ 10^{-3}$	$6.0\ 10^{-3}$	$6.2\ 10^{-3}$	$1.0\ 10^{-2}$	$4.0\ 10^{-2}$	$4.2\ 10^{-2}$
^{236}U	$2.4\ 10^7$	$1.0\ 10^{-1}$	$1.0\ 10^{-1}$	$1.5\ 10^{-1}$	$4.1\ 10^{-1}$	$5.9\ 10^{-1}$	$5.7\ 10^{-1}$
^{238}U	$4.5\ 10^9$	$1.2\ 10^{-1}$	$1.2\ 10^{-1}$	$1.2\ 10^{-1}$	$1.2\ 10^{-1}$	$1.2\ 10^{-1}$	$1.2\ 10^{-1}$
^{237}Np	$2.1\ 10^6$	$1.5\ 10^1$	$1.5\ 10^1$	$1.8\ 10^1$	$1.9\ 10^1$	$1.8\ 10^1$	$1.4\ 10^1$
^{238}Pu	$8.9\ 10^1$	$2.2\ 10^3$	$1.6\ 10^3$	$2.8\ 10^1$	-	-	-
^{239}Pu	$2.4\ 10^4$	$1.8\ 10^2$	$1.9\ 10^2$	$2.4\ 10^2$	$5.9\ 10^2$	$8.4\ 10^1$	-
^{240}Pu	$6.8\ 10^3$	$2.0\ 10^2$	$1.7\ 10^3$	$1.6\ 10^3$	$6.5\ 10^2$	$7.4\ 10^{-2}$	-
^{241}Pu	$1.5\ 10^1$	$4.4\ 10^4$	$8.2\ 10^2$	$1.4\ 10^2$	$6.8\ 10^1$	$3.7\ 10^{-2}$	-
^{242}Pu	$3.8\ 10^5$	$1.0\ 10^0$	$1.0\ 10^0$	$1.1\ 10^0$	$1.1\ 10^0$	$9.1\ 10^{-1}$	$1.8\ 10^{-1}$
^{241}Am	$4.3\ 10^2$	$1.7\ 10^4$	$1.6\ 10^4$	$3.8\ 10^3$	$6.8\ 10^1$	$3.9\ 10^{-2}$	-
^{243}Am	$7.7\ 10^3$	$2.8\ 10^3$	$2.8\ 10^3$	$2.6\ 10^3$	$1.1\ 10^3$	$3.5\ 10^{-1}$	-

a/ Short-lived daughters in secular equilibrium with their parent-nuclides have not been included in the table.

Table 2.1

Activity composition of relevant actinides and fission products in high level waste as a function of time, normalized per kWh electrical energy, for fuel cycle I Fast Breeder Reactor with plutonium recycle

Nuclide[a]	Half-life (y)	Time after removal from reactor: 10^1 y	10^2 y	10^3 y	10^4 y	10^5 y	10^6 y
		Normalized activity (Ci per GW(e)y)					
^{99}Tc	$2.1 \cdot 10^5$	$4.5 \cdot 10^2$	$4.5 \cdot 10^2$	$4.5 \cdot 10^2$	$4.4 \cdot 10^2$	$3.2 \cdot 10^2$	$1.7 \cdot 10^1$
^{129}I	$1.7 \cdot 10^7$	$8.4 \cdot 10^{-1}$	$8.4 \cdot 10^{-1}$	$8.4 \cdot 10^{-1}$	$8.4 \cdot 10^{-1}$	$8.4 \cdot 10^{-1}$	$8.1 \cdot 10^{-1}$
^{135}Cs	$3.0 \cdot 10^6$	$3.4 \cdot 10^1$	$3.4 \cdot 10^1$	$3.4 \cdot 10^1$	$3.4 \cdot 10^1$	$3.3 \cdot 10^1$	$2.7 \cdot 10^1$
^{226}Ra	$1.6 \cdot 10^3$	-	-	$2.2 \cdot 10^{-3}$	$3.6 \cdot 10^{-1}$	$2.8 \cdot 10^0$	$6.0 \cdot 10^{-1}$
^{234}U	$2.5 \cdot 10^5$	$2.0 \cdot 10^{-1}$	$2.3 \cdot 10^0$	$5.8 \cdot 10^0$	$5.7 \cdot 10^0$	$4.5 \cdot 10^0$	$4.3 \cdot 10^{-1}$
^{235}U	$7.1 \cdot 10^8$	$8.0 \cdot 10^{-4}$	$9.1 \cdot 10^{-4}$	$2.0 \cdot 10^{-3}$	$1.2 \cdot 10^{-2}$	$5.7 \cdot 10^{-2}$	$6.1 \cdot 10^{-2}$
^{236}U	$2.4 \cdot 10^7$	$2.6 \cdot 10^{-3}$	$7.4 \cdot 10^{-3}$	$5.4 \cdot 10^{-2}$	$3.4 \cdot 10^{-1}$	$5.4 \cdot 10^{-1}$	$5.2 \cdot 10^{-1}$
^{238}U	$4.5 \cdot 10^9$	$7.0 \cdot 10^{-2}$	$7.0 \cdot 10^{-2}$	$7.0 \cdot 10^{-2}$	$7.0 \cdot 10^{-2}$	$7.0 \cdot 10^{-2}$	$7.0 \cdot 10^{-2}$
^{237}Np	$2.1 \cdot 10^6$	$3.9 \cdot 10^0$	$6.1 \cdot 10^0$	$1.7 \cdot 10^1$	$2.1 \cdot 10^1$	$2.0 \cdot 10^1$	$1.5 \cdot 10^1$
^{238}Pu	$8.9 \cdot 10^1$	$1.1 \cdot 10^4$	$6.6 \cdot 10^3$	$7.8 \cdot 10^1$	-	-	-
^{239}Pu	$2.4 \cdot 10^4$	$1.3 \cdot 10^3$	$1.3 \cdot 10^3$	$1.3 \cdot 10^3$	$1.2 \cdot 10^3$	$1.1 \cdot 10^2$	-
^{240}Pu	$6.8 \cdot 10^3$	$1.8 \cdot 10^3$	$1.8 \cdot 10^3$	$1.6 \cdot 10^3$	$6.8 \cdot 10^2$	$7.1 \cdot 10^{-2}$	-
^{241}Pu	$1.5 \cdot 10^1$	$1.3 \cdot 10^5$	$2.1 \cdot 10^3$	$1.1 \cdot 10^3$	$5.1 \cdot 10^2$	-	-
^{242}Pu	$3.8 \cdot 10^5$	$5.3 \cdot 10^0$	$5.4 \cdot 10^0$	$5.5 \cdot 10^0$	$5.5 \cdot 10^0$	$4.6 \cdot 10^0$	$9.0 \cdot 10^{-1}$
^{241}Am	$4.3 \cdot 10^2$	$7.7 \cdot 10^4$	$1.2 \cdot 10^5$	$3.1 \cdot 10^4$	$9.5 \cdot 10^{-2}$	-	-
^{243}Am	$7.7 \cdot 10^3$	$1.5 \cdot 10^3$	$1.5 \cdot 10^3$	$1.4 \cdot 10^3$	$6.1 \cdot 10^2$	$1.9 \cdot 10^{-1}$	-

a/ Short-lived daughters in secular equilibrium with their parent-nuclides have not been included in the table.

Table 3.4

Activity composition of relevant actinides and fission products in discarded spent fuel as a function of time, normalized per unit electrical energy, for Fuel Cycle 4 Heavy Water Reactor with spent fuel disposal

Nuclide[a]	Half-life (y)	Time after removal from reactor					
		10^{1} y	10^{2} y	10^{3} y	10^{4} y	10^{5} y	10^{6} y
		Normalized activity (Ci per GW(e))					
^{99}Tc	$2.1\ 10^{5}$	$2.1\ 10^{2}$	$2.1\ 10^{2}$	$2.1\ 10^{2}$	$2.0\ 10^{2}$	$1.5\ 10^{2}$	$7.7\ 10^{0}$
^{129}I	$1.7\ 10^{7}$	$1.2\ 10^{0}$	$1.2\ 10^{0}$	$1.2\ 10^{0}$	$1.2\ 10^{0}$	$1.2\ 10^{0}$	$1.2\ 10^{0}$
^{135}Cs	$3.0\ 10^{6}$	$3.8\ 10^{0}$	$3.8\ 10^{0}$	$3.8\ 10^{0}$	$3.8\ 10^{0}$	$3.7\ 10^{0}$	$3.0\ 10^{0}$
^{226}Ra	$1.6\ 10^{3}$	-	-	$8.5\ 10^{-2}$	$3.4\ 10^{0}$	$3.1\ 10^{1}$	$5.8\ 10^{1}$
^{234}U	$2.5\ 10^{5}$	$5.1\ 10^{1}$	$5.3\ 10^{1}$	$5.4\ 10^{1}$	$5.4\ 10^{1}$	$5.5\ 10^{1}$	$5.9\ 10^{1}$
^{235}U	$7.1\ 10^{8}$	$9.3\ 10^{-1}$	$9.3\ 10^{-1}$	$9.6\ 10^{-1}$	$1.2\ 10^{0}$	$1.8\ 10^{0}$	$1.9\ 10^{0}$
^{236}U	$2.4\ 10^{7}$	$8.2\ 10^{0}$	$8.3\ 10^{0}$	$9.2\ 10^{0}$	$1.5\ 10^{1}$	$1.8\ 10^{1}$	$1.8\ 10^{1}$
^{238}U	$4.5\ 10^{9}$	$5.9\ 10^{1}$	$5.9\ 10^{1}$	$5.9\ 10^{1}$	$5.9\ 10^{1}$	$5.9\ 10^{1}$	$5.9\ 10^{1}$
^{237}Np	$2.1\ 10^{6}$	$3.4\ 10^{0}$	$5.8\ 10^{0}$	$2.0\ 10^{1}$	$2.4\ 10^{1}$	$2.4\ 10^{1}$	$1.8\ 10^{1}$
^{238}Pu	$8.9\ 10^{1}$	$9.0\ 10^{3}$	$4.5\ 10^{3}$	$4.5\ 10^{1}$	-	-	-
^{239}Pu	$2.4\ 10^{4}$	$2.9\ 10^{4}$	$2.9\ 10^{4}$	$2.8\ 10^{4}$	$2.2\ 10^{4}$	$1.6\ 10^{3}$	-
^{240}Pu	$6.8\ 10^{3}$	$3.8\ 10^{4}$	$3.8\ 10^{4}$	$3.4\ 10^{4}$	$1.4\ 10^{3}$	$1.4\ 10^{0}$	-
^{241}Pu	$1.5\ 10^{1}$	$1.9\ 10^{6}$	$3.1\ 10^{4}$	$3.6\ 10^{-2}$	$1.7\ 10^{-2}$	-	-
^{242}Pu	$3.8\ 10^{5}$	$3.1\ 10^{1}$	$3.1\ 10^{1}$	$3.1\ 10^{1}$	$3.0\ 10^{1}$	$2.6\ 10^{1}$	$5.0\ 10^{0}$
^{241}Am	$4.3\ 10^{2}$	$3.9\ 10^{4}$	$9.0\ 10^{4}$	$2.1\ 10^{4}$	$1.1\ 10^{-2}$	-	-
^{243}Am	$7.7\ 10^{3}$	$6.3\ 10^{1}$	$6.2\ 10^{1}$	$5.8\ 10^{1}$	$2.6\ 10^{1}$	$7.8\ 10^{-3}$	-

a/ Short-lived daughters in secular equilibrium with their parent-nuclides have not been included in the table.

Table 2.5

Activity composition of relevant actinides and fission products in high level waste as a function of time, normalized per unit electrical energy, for Fuel Cycle 5 Heavy Water Reactor with plutonium recycle

Nuclide[a]	Half-life (y)	Time after removal from reactor					
		10^1 y	10^2 y	10^3 y	10^4 y	10^5 y	10^6 y
		Normalized activity (Ci per GW(e)y)					
^{99}Tc	$2.1\ 10^5$	$5.4\ 10^2$	$5.4\ 10^2$	$5.4\ 10^2$	$5.2\ 10^2$	$3.9\ 10^2$	$2.0\ 10^1$
^{129}I	$1.7\ 10^7$	-	-	-	-	-	-
^{135}Cs	$3.0\ 10^6$	$5.0\ 10^0$	$5.0\ 10^0$	$5.0\ 10^0$	$5.0\ 10^0$	$4.9\ 10^0$	$4.0\ 10^0$
^{226}Ra	$1.6\ 10^3$	-	-	-	$2.6\ 10^{-2}$	$2.2\ 10^{-1}$	$2.6\ 10^{-1}$
^{234}U	$2.5\ 10^5$	$2.0\ 10^{-1}$	$2.8\ 10^{-1}$	$4.2\ 10^{-1}$	$4.2\ 10^{-1}$	$3.8\ 10^{-1}$	$2.6\ 10^{-1}$
^{235}U	$7.1\ 10^8$	$2.2\ 10^{-3}$	$2.2\ 10^{-3}$	$2.4\ 10^{-3}$	$1.2\ 10^{-2}$	$9.2\ 10^{-2}$	$1.0\ 10^{-1}$
^{236}U	$2.4\ 10^7$	$1.5\ 10^{-2}$	$2.1\ 10^{-2}$	$9.3\ 10^{-2}$	$5.4\ 10^{-1}$	$8.4\ 10^{-1}$	$8.2\ 10^{-1}$
^{238}U	$4.5\ 10^9$	$2.5\ 10^{-1}$	$2.5\ 10^{-1}$	$2.5\ 10^{-1}$	$2.5\ 10^{-1}$	$2.5\ 10^{-1}$	$2.5\ 10^{-1}$
^{237}Np	$2.1\ 10^6$	$3.7\ 10^0$	$5.9\ 10^0$	$1.7\ 10^1$	$2.0\ 10^1$	$2.0\ 10^1$	$1.5\ 10^1$
^{238}Pu	$8.9\ 10^1$	$4.4\ 10^2$	$2.6\ 10^2$	$2.9\ 10^0$	-	-	-
^{239}Pu	$2.4\ 10^4$	$1.4\ 10^2$	$1.6\ 10^2$	$3.7\ 10^2$	$1.5\ 10^3$	$2.3\ 10^2$	-
^{240}Pu	$6.8\ 10^3$	$5.8\ 10^2$	$2.8\ 10^3$	$2.6\ 10^3$	$1.1\ 10^3$	$1.1\ 10^{-1}$	-
^{241}Pu	$1.5\ 10^1$	$3.8\ 10^4$	$5.6\ 10^2$	$8.3\ 10^0$	$3.6\ 10^0$	$2.2\ 10^{-3}$	-
^{242}Pu	$3.8\ 10^5$	$7.3\ 10^0$	$7.3\ 10^0$	$7.3\ 10^0$	$7.2\ 10^0$	$6.1\ 10^0$	$1.2\ 10^0$
^{241}Am	$4.3\ 10^2$	$8.0\ 10^4$	$7.0\ 10^4$	$1.6\ 10^4$	$9.2\ 10^{-3}$	-	-
^{243}Am	$7.7\ 10^3$	$8.6\ 10^3$	$8.5\ 10^3$	$7.9\ 10^3$	$3.5\ 10^3$	$1.1\ 10^0$	-

a/ Short-lived daughters in secular equilibrium with their parent-nuclides have not been included in the table.

Table 3.1

Activity composition of relevant actinides and fission products in high level waste as a function of time, normalized per unit electrical energy, for fuel cycle 6 Heavy Water Reactor with uranium-thorium recycle

Nuclide[a]	Half-life (y)	Time after removal from reactor					
		10^1 y	10^2 y	10^3 y	10^4 y	10^5 y	10^6 y
		Normalized activity (Ci per GW(e)y)					
^{99}Tc	$2.1\ 10^5$	$1.7\ 10^2$	$1.7\ 10^2$	$1.7\ 10^2$	$1.6\ 10^2$	$1.2\ 10^2$	$6.3\ 10^0$
^{129}I	$1.7\ 10^7$	-	-	-	-	-	-
^{135}Cs	$3.0\ 10^6$	$6.2\ 10^0$	$6.2\ 10^0$	$6.2\ 10^0$	$6.2\ 10^0$	$6.1\ 10^0$	$4.9\ 10^0$
^{226}Ra	$1.6\ 10^3$	-	-	$8.5\ 10^{-2}$	$3.4\ 10^0$	$2.7\ 10^1$	$5.0\ 10^0$
^{225}Th	$1.9\ 10^0$	$1.9\ 10^2$	$1.3\ 10^{-1}$	$1.3\ 10^{-1}$	$1.3\ 10^{-1}$	$1.3\ 10^{-1}$	$1.3\ 10^{-1}$
^{229}Th	$7.3\ 10^3$	$1.8\ 10^{-1}$	$1.7\ 10^0$	$1.6\ 10^1$	$1.1\ 10^2$	$1.3\ 10^2$	$8.5\ 10^0$
^{230}Th	$8.0\ 10^4$	$4.4\ 10^{-3}$	$4.2\ 10^{-2}$	$4.6\ 10^{-1}$	$4.4\ 10^0$	$2.7\ 10^1$	$5.0\ 10^0$
^{232}Th	$1.4\ 10^{10}$	$1.3\ 10^{-1}$	$1.3\ 10^{-1}$	$1.3\ 10^{-1}$	$1.3\ 10^{-1}$	$1.3\ 10^{-1}$	$1.3\ 10^{-1}$
^{232}U	$7.2\ 10^1$	-	-	-	-	-	-
^{233}U	$1.6\ 10^5$	$1.8\ 10^2$	$1.8\ 10^2$	$1.8\ 10^2$	$1.7\ 10^2$	$1.3\ 10^2$	$8.5\ 10^0$
^{234}U	$2.5\ 10^5$	$4.6\ 10^1$	$5.0\ 10^1$	$5.4\ 10^1$	$5.3\ 10^1$	$4.1\ 10^1$	$3.4\ 10^0$
^{235}U	$7.1\ 10^8$	$3.1\ 10^{-3}$	$3.1\ 10^{-3}$	$3.6\ 10^{-3}$	$7.2\ 10^{-3}$	$1.9\ 10^{-2}$	$2.0\ 10^{-2}$
^{236}U	$2.4\ 10^7$	$2.4\ 10^{-1}$	$2.4\ 10^{-1}$	$2.5\ 10^{-1}$	$3.2\ 10^{-1}$	$3.7\ 10^{-1}$	$3.6\ 10^{-1}$
^{238}U	$4.5\ 10^9$	$2.4\ 10^{-3}$	$2.4\ 10^{-3}$	$2.4\ 10^{-3}$	$2.4\ 10^{-3}$	$2.4\ 10^{-3}$	$2.4\ 10^{-3}$
^{237}Np	$2.1\ 10^6$	$8.6\ 10^0$	$8.6\ 10^0$	$9.0\ 10^0$	$9.1\ 10^0$	$8.8\ 10^0$	$6.7\ 10^0$
^{238}Pu	$8.9\ 10^1$	$2.3\ 10^4$	$1.1\ 10^4$	$1.1\ 10^1$	-	-	-
^{239}Pu	$2.4\ 10^4$	$4.9\ 10^2$	$4.9\ 10^2$	$4.8\ 10^2$	$3.7\ 10^2$	$2.7\ 10^1$	-
^{240}Pu	$6.8\ 10^3$	$4.6\ 10^2$	$4.6\ 10^2$	$4.2\ 10^2$	$1.7\ 10^2$	$1.7\ 10^{-2}$	-
^{241}Pu	$1.5\ 10^1$	$4.4\ 10^4$	$6.9\ 10^2$	$8.7\ 10^0$	$4.1\ 10^0$	$2.2\ 10^{-3}$	-
^{242}Pu	$3.8\ 10^5$	$5.2\ 10^{-1}$	$5.2\ 10^{-1}$	$5.2\ 10^{-1}$	$5.1\ 10^{-1}$	$4.3\ 10^{-1}$	$8.4\ 10^{-2}$
^{241}Am	$4.3\ 10^2$	$9.1\ 10^2$	$2.3\ 10^3$	$5.4\ 10^2$	$2.7\ 10^{-4}$	-	-
^{243}Am	$7.7\ 10^3$	$1.1\ 10^0$	$1.1\ 10^0$	$1.0\ 10^0$	$4.5\ 10^{-1}$	-	-

a/ Short-lived daughters in secular equilibrium with their parent-nuclides have not been included in the table.

Table 2.7

Activity composition of relevant actinides and fission products in high level waste as a function of time, normalized per unit electrical energy, for Fuel Cycle 7 High Temperature Reactor with uranium-thorium recycle

Nuclide[a]	Half-life (y)	Time after removal from reactor					
		10^1 y	10^2 y	10^3 y	10^4 y	10^5 y	10^6 y
		Normalized activity (Ci per GW(e)y)					
^{99}Tc	2.1 10^5	4.9 10^1	4.9 10^1	4.9 10^1	4.7 10^1	3.5 10^1	1.8 10^0
^{129}I	1.7 10^7	7.0 10^{-1}	7.0 10^{-1}	7.0 10^{-1}	7.0 10^{-1}	7.0 10^{-1}	6.7 10^{-1}
^{135}Cs	3.0 10^6	4.7 10^0	4.7 10^0	4.7 10^0	4.7 10^0	4.6 10^0	3.7 10^0
^{226}Ra	1.6 10^3	-	4.0 10^{-3}	8.0 10^{-2}	2.2 10^0	1.7 10^{-1}	3.1 10^0
^{228}Th	1.9 10^0	4.0 10^1	1.9 10^1	1.8 10^{-2}	1.5 10^{-2}	1.5 10^{-2}	1.5 10^{-2}
^{229}Th	7.3 10^3	3.6 10^{-2}	3.4 10^{-1}	3.3 10^0	2.2 10^1	2.7 10^1	7.2 10^0
^{230}Th	8.0 10^4	1.0 10^{-1}	1.1 10^{-1}	3.6 10^{-1}	2.8 10^0	1.7 10^1	3.1 10^0
^{232}Th	1.4 10^{10}	1.5 10^{-2}	1.5 10^{-2}	1.5 10^{-2}	1.5 10^{-2}	1.5 10^{-2}	1.5 10^{-2}
^{232}U	7.2 10^1	4.3 10^1	1.8 10^1	3.1 10^{-3}	-	-	-
^{233}U	1.6 10^5	3.6 10^1	3.6 10^1	3.6 10^1	3.5 10^1	2.6 10^1	7.2 10^0
^{234}U	2.5 10^5	9.4 10^0	2.2 10^1	3.4 10^1	3.3 10^1	2.5 10^1	2.1 10^0
^{235}U	7.1 10^8	1.8 10^{-2}	1.8 10^{-2}	1.8 10^{-2}	1.8 10^{-2}	2.0 10^{-2}	2.0 10^{-2}
^{236}U	2.4 10^7	3.0 10^0	3.0 10^0	3.0 10^0	3.0 10^0	3.0 10^0	2.9 10^0
^{238}U	4.5 10^9	4.3 10^{-3}	4.3 10^{-3}	4.3 10^{-3}	4.3 10^{-3}	4.3 10^{-3}	4.3 10^{-3}
^{237}Np	2.1 10^6	8.5 10^0	8.5 10^0	8.7 10^0	8.8 10^0	8.5 10^0	6.3 10^0
^{238}Pu	8.9 10^1	6.8 10^4	3.4 10^4	3.1 10^1	-	-	-
^{239}Pu	2.4 10^4	5.6 10^1	5.6 10^1	5.5 10^1	4.6 10^1	3.7 10^0	-
^{240}Pu	6.8 10^3	9.5 10^1	1.0 10^2	9.3 10^1	3.7 10^1	3.9 10^{-3}	-
^{241}Pu	1.5 10^1	2.6 10^4	3.6 10^2	-	-	-	-
^{242}Pu	3.8 10^5	1.6 10^0	1.6 10^0	1.6 10^0	1.6 10^0	1.3 10^0	2.6 10^{-1}
^{241}Am	4.3 10^2	5.6 10^2	1.3 10^3	3.0 10^2	-	-	-
^{243}Am	7.7 10^3	2.4 10^1	2.4 10^1	2.2 10^1	1.0 10^1	3.0 10^{-3}	-

a/ Short-lived daughters in secular equilibrium with their parent-nuclides have not been included in the table.

50. As only a range of possible values of the collective dose commitment will be estimated (paragraph 43), the collective dose commitments will be assessed as a function only of the "age" of the waste mixture when it reaches surface waters, assuming no other change of the waste nuclide composition than that caused by decay or by genetic generation through a decay chain.

51. For comparison, collective dose commitments will also be calculated for the estimated arrival of radionuclides from a disrupted salt repository (paragraph 44 to 46). Table 2.a summarizes the input of radionuclides into fresh waters, estimated by a transport model (72). This impact covers not only that from high level waste but also from wastes from enrichment, fuel fabrication and from depleted uranium.

4.3.1.1. Fresh Water Contribution

52. Waste radionuclides entering fresh waters result in doses to man mainly through the pathways of drinking water and fish consumption. If irrigation of crops is practiced, specially by sprinkling methods, this route could contribute significantly to the collective dose. Other fresh water pathways have generally been found to contribute negligibly to the collective dose (1,44) even if they could be limiting, in some cases, considering individual doses.

53. The collective effective dose equivalent commitment from the input of radionuclides in fresh waters, normalized per GW(e)y, can be estimated (1), using the expression

$$S^{C}_{E,i} = \frac{1}{V} \sum_{i} \frac{A_{i,1}}{\lambda_i + \frac{1}{\tau}} \sum_{k} N_k \, T_k \, f_{k,i} \, F_i$$

Table 2.a

Input of Radionuclides into Circulating Waters from a Salt Repository, Assessed by a Transport Model (72)

Nuclide	Strategy						
	1	2	3	4	5	6	7
	Normalized activity (Ci per GW(e)y)						
^{99}Tc	$3.4\ 10^{-1}$	$3.4\ 10^{-1}$	$3.1\ 10^{-1}$	$3.7\ 10^{-1}$	$3.7\ 10^{-1}$	$3.0\ 10^{-1}$	$2.2\ 10^{-2}$
^{129}I	$1.3\ 10^{0}$	$1.3\ 10^{0}$	$8.2\ 10^{-1}$	$1.2\ 10^{0}$	$1.3\ 10^{0}$	$1.9\ 10^{0}$	$1.2\ 10^{0}$
^{135}Cs	$5.6\ 10^{0}$	$8.3\ 10^{0}$	$2.3\ 10^{1}$	$2.6\ 10^{0}$	$3.4\ 10^{0}$	$4.3\ 10^{0}$	$3.2\ 10^{0}$
^{226}Ra	$1.4\ 10^{2}$	$9.1\ 10^{1}$	$2.5\ 10^{-1}$	$1.3\ 10^{2}$	$5.6\ 10^{1}$	$5.2\ 10^{0}$	$4.1\ 10^{1}$
^{231}Pa	$5.2\ 10^{-2}$	$1.8\ 10^{-2}$	$1.9\ 10^{-3}$	$6.6\ 10^{-2}$	$1.1\ 10^{-2}$	$1.5\ 10^{-3}$	$8.7\ 10^{-3}$
^{229}Th	$5.3\ 10^{-3}$	$1.9\ 10^{-3}$	$2.8\ 10^{-3}$	$2.4\ 10^{-3}$	$2.2.10^{-3}$	$5.3\ 10^{-4}$	$9.3\ 10^{-4}$
^{230}Th	$2.9\ 10^{0}$	$1.8\ 10^{0}$	$4.7\ 10^{-3}$	$2.6\ 10^{0}$	$1.1\ 10^{0}$	$9.3\ 10^{-2}$	$7.9\ 10^{-1}$
^{232}Th	$2.2\ 10^{-3}$	$9.0\ 10^{-4}$	$9.3\ 10^{-5}$	$3.0\ 10^{-3}$	$3.8\ 10^{-4}$	$1.4\ 10^{-1}$	$1.6\ 10^{-2}$
^{233}U	$1.5\ 10^{-1}$	$7.4\ 10^{-2}$	$8.1\ 10^{-2}$	$9.5\ 10^{-2}$	$8.1\ 10^{-2}$	$3.6\ 10^{-2}$	$3.5\ 10^{-2}$
^{234}U	$6.7\ 10^{1}$	$4.1\ 10^{1}$	$1.1\ 10^{-1}$	$6.1\ 10^{1}$	$2.6\ 10^{1}$	$2.3\ 10^{0}$	$1.9\ 10^{1}$
^{235}U	$1.7\ 10^{0}$	$5.3\ 10^{-1}$	$6.5\ 10^{-2}$	$2.0\ 10^{0}$	$3.2\ 10^{-1}$	$4.7\ 10^{-2}$	$2.6\ 10^{-1}$
^{236}U	$1.3\ 10^{1}$	$5.1\ 10^{0}$	$5.3\ 10^{-1}$	$1.7\ 10^{1}$	$2.1\ 10^{0}$	$3.5\ 10^{-1}$	$2.8\ 10^{0}$
^{238}U	$6.6\ 10^{1}$	$4.0\ 10^{1}$	$1.1\ 10^{-1}$	$6.0\ 10^{1}$	$2.5\ 10^{1}$	$2.3\ 10^{0}$	$1.9\ 10^{1}$
^{237}Np	$8.4\ 10^{-3}$	$4.1\ 10^{-3}$	$4.5\ 10^{-3}$	$5.3\ 10^{-3}$	$4.5\ 10^{-3}$	$2.0\ 10^{-3}$	$2.0\ 10^{-3}$

where $A_{i,1}$ is the activity input of nuclide i, per GW(e)y in the receiving fresh water

V is the volume of receiving waters

τ is the turnover time of receiving waters

λ_i is the decay constant of nuclide i

N_k is the number of individuals exposed by pathway k

I_k is the individual consumption rate of pathway item k

$f_{k,i}$ is the concentration factor for the consumed item in pathway k for nuclide i

F_i is the collective effective dose equivalent per unit activity ingested collectively by the exposed group.

54. The quantity $\frac{A_{i,1}}{V\,(\lambda_i + \frac{1}{\tau})}$ is the infinite-time integral of the water concentration of nuclide i, while the quantity multiplied by $f_{k,i}$ is the infinite-time integral of the concentration of nuclide i in the consumed item k. For radionuclide inputs into small volumes of water, the concentrations in water and in fish will be high, but the population which can be served with drinking water or by fish consumption will be limited. For inputs into larger volumes of water, the concentrations will be smaller, but the population involved will be correspondingly larger.

55. It seems reasonable to assume, therefore, as a first approximation, that the quantities $\frac{N_k\ I_k}{V}$ are relatively constant. Representative values of these quantities can be estimated as global averages or for regional areas. Both procedures give similar results. For example,

for the drinking pathway, a global estimate of $\frac{N_k}{V} \approx 3\ 10^{-8}\ \frac{man}{l_9}$ can be obtained from the world population ($3.8\ 10^9$) and the world availability for fresh waters ($1.3\ 10^{17}$kg) (45). Some regional estimates range within an order of magnitude of the global estimate (46, 47, 48). Using the global estimate, and assuming a per caput intake rate of water of 2.5 kg per day, the quantity $\frac{N_k\ I_k}{V}$ for drinking water is estimated to be $2.7\ 10^{-5}\ y^{-1}$.

56. By a similar procedure it is possible to estimate $\frac{N_k\ I_k}{V}$ for fish consumption. Total fresh water fish consumption by the world population is about $3.8\ 10^9$ kg y^{-1} (49), which agrees well with the estimated annual global catch of 10^{10}kg (45), taking into account a correction for edible weight. Dividing this global consumption rate by the world availability of fresh waters discussed in paragraph 55, a value of $\frac{N_k\ I_k}{V} \approx 3\ 10^{-8} y^{-1}$ for the fish consumption pathway is obtained.

57. In the calculation of collective dose commitments contributions using the formula of paragraph 53, a global fresh-water turnover time of 10 years has been assumed (1). Sediment removal has been neglected due to the short time scale involved. Table 3 sumarizes the other factors required for the assessment of the collective dose commitment contributions, together with the resulting collective dose commitments per unit input of several relevant nuclides in fresh waters.

58. Concentration factors for drinking water (activity removal by purification) and for fish shown in Table 3 are partly taken from available compilations (47, 50, 51).

Table 3

Collective dose commitment factors
for the input of radionuclides in fresh waters

Nuclide	f_w (1)	f_f (2)	F_i (3) (man rem/Ci)
^{99}Tc	0.8	15	$5.5\ 10^2$
^{129}I	0.8	15	$3.4\ 10^5$
^{135}Cs	0.2	400	$7.3\ 10^3$
^{226}Ra	0.5	4	$2.8\ 10^6$
^{229}Th	0.1	10	$3.4\ 10^5$
^{230}Th	0.1	10	$3.4\ 10^5$
^{233}U	0.1	10	$1.1\ 10^5$
^{234}U	0.1	10	$1.1\ 10^5$
^{235}U	0.1	10	$1.1\ 10^5$
^{236}U	0.1	10	$1.1\ 10^5$
^{238}U	0.1	10	$1.1\ 10^5$
^{237}Np	0.1	10	$2.0\ 10^5$
^{239}Pu	0.1	10	$1.6\ 10^5$
^{240}Pu	0.1	10	$1.6\ 10^5$
^{241}Am	0.1	10	$2.2\ 10^5$
^{243}Am	0.1	10	$2.2\ 10^5$

Notes:

(1) f_w is the concentration factor for drinking water

(2) f_f is the concentration factor for fish

(3) F_i is the collective effective dose equivalent per unit activity ingested collectively by the exposed group.

As values are unavailable for most actinides, it has been assumed that the cerium values are applicable in those cases. Radium values are derived from studies on releases from uranium mills (52). The dosimetric conversion factors F_i have been derived applying the organ weighting factors of ICRP publication 26 (2) to the dosimetric models of ICRP publication 2 (53), except for ^{226}Ra for which a more recent dosimetric model has been used (1).

4.3.1.2. Ocean Contribution

59. A basic assumption of the assessments presented in this report is that the waste nuclides will follow ground water, with some delay, and will seep into fresh waters finally flowing into the sea. In view of the time scale of the residence of the nuclides in fresh waters and the very long lifes involved, it can be assumed that the input into the sea will contain the entire radionuclide inventory that had been introduced into fresh waters.

60. Any model selected to describe the dispersion of radionuclides in the ocean would be a vast simplification of the real situation. Due to the very long-lived materials involved, and the fact that the collective dose commitment is the desired quantity, it is possible to neglect from the assessment the initial dispersion, until a relatively uniform concentration is achieved in all ocean compartments. This initial dispersion would require approximately one thousand years (54).

61. In these conditions, the collective dose commitment contribution from all exposure pathways starting with sea water will be proportional to the infinite-time integral of the water concentration, which in turn depends on the mean life of the nuclide and on the rate of its removal from the water.

62. Many nuclides dispersed in the marine environment become associated with sediments. The capacity of sediments to remove nuclides depend on several physical and chemical parameters. Experimental and field determinations of the distribution coefficient for many nuclides and marine sediment types have been carried out (55, 56, 57, 58, 59, 60). When the radionuclides have reached uniform distribution, the rate of their removal from the water can be calculated from the distribution coefficient by the relation (54)

$$S = \frac{K\ v}{h}$$

where S is the fractional rate of removal, K is the distribution coefficient, v is the velocity of sedimentation and h is the average depth of the water column. Taking a mean sedimentation velocity of 10^{-5} m/y (61, 62) and a mean of ocean depth of about 4000 m, the removal rate can be expressed as $S = 2.5\ 10^{-9}\ y^{-1}\ K$.

63. If sedimentation and radioactive decay are the only removal processes operating on waste nuclides, then the infinite-time integral of the water concentration is given by

$$\int_0^\infty C_i(t)\ dt = \frac{A_i}{V\ (\lambda_i + S_i)}$$

where $C_i(t)$ is the concentration of nuclide i in water, A_i is the total input of that nuclide into the oceans, V is the volume of the diluting waters ($4.6\ 10^{17}\ m^3$), λ_i is the decay constant of nuclide i, and S_i is the removal rate of nuclide i by sedimentation.

64. Contrary to the fresh-water case, where the short turnover time made the contribution of long-lived daughter nuclides insignificant, the assessment of collective dose commitments mediated through the oceans requires that the parent-daughter relationships for the nuclides are taken into account. A further complication derives from the fact that some daughter nuclides are less retained by the sediments than the parent nuclides. In these cases the sediments become an input source for the daughter nuclide.

65. Several pathways should be considered in the assessment of human exposures from radionuclides dispersed into the oceans, e.g. consumption of fish (and sea foods), sediment resuspension and inhalation, and external exposure to sediments. Potential future pathways should also be considered, such as consumption of fresh water obtained by desalting, plankton, etc.

66. Table 4 sumarizes the transfer factors used in this report to assess the collective dose contributions of the present exposure pathways. The concentration factors for fish (38, 63, 64, 65, 66, 67) are applicable to the edible parts and expressed as ratios by weight. The distribution coefficients for sediments (paragraph 62), on the other hand, are expressed as ratios by volume. The dosimetric conversion factors F_i (ingestion) and F_a (inhalation from a "cloud exposure") have been derived applying the organ weighting factors of ICRP publication 26 (2) to the dosimetric models of ICRP publications 2 (53), except for ^{226}Ra for which recently published dosimetric models have been used (1).

67. The collective dose commitment contribution of the fish consumption pathway can be calculated from the time-integrals of the water concentrations using the relevant

Table 4

Collective dose commitment factors

for the input of radionuclides into the oceans

Nuclide	λ (y^{-1})	f_f (1)	K (2)	F_i (3) (man rem/Ci)	F_a (4) (rem/Ci y m^{-3})
^{99}Tc	$3.3\ 10^{-6}$	10	10^4	$5.5\ 10^2$	$2.6\ 10^6$
^{129}I	$4.1\ 10^{-8}$	10	10^4	$3.4\ 10^5$	$1.4\ 10^9$
^{135}Cs	$3.5\ 10^{-7}$	30	10^2	$7.3\ 10^3$	$4.2\ 10^7$
^{226}Ra	$4.4\ 10^{-4}$	100	10^5	$2.8\ 10^4$	$2.8\ 10^{10}$
^{229}Th	$9.5\ 10^{-5}$	10^4	10^7	$3.4\ 10^5$	$6.6\ 10^{12}$
^{230}Th	$8.8\ 10^{-6}$	10^4	10^7	$3.4\ 10^5$	$6.6\ 10^{12}$
^{232}U	$4.4\ 10^{-6}$	10	10^4	$1.1\ 10^5$	$2.0\ 10^{10}$
^{234}U	$2.8\ 10^{-6}$	10	10^4	$1.1\ 10^5$	$2.0\ 10^{10}$
^{235}U	$9.9\ 10^{-10}$	10	10^4	$1.0\ 10^5$	$1.9\ 10^{10}$
^{236}U	$3.0\ 10^{-8}$	10	10^4	$1.0\ 10^5$	$1.9\ 10^{10}$
^{238}U	$1.5\ 10^{-10}$	10	10^4	$1.0\ 10^5$	$1.9\ 10^{10}$
^{237}Np	$3.3\ 10^{-7}$	10	10^4	$2.0\ 10^5$	$3.7\ 10^{12}$
^{238}Pu	$7.9\ 10^{-3}$	10	$5\ 10^4$	$1.6\ 10^5$	$6.9\ 10^{12}$
^{239}Pu	$2.9\ 10^{-5}$	10	$5\ 10^4$	$1.6\ 10^5$	$6.9\ 10^{12}$
^{240}Pu	$1.1\ 10^{-4}$	10	$5\ 10^4$	$1.6\ 10^5$	$6.9\ 10^{12}$
^{242}Pu	$1.9\ 10^{-6}$	10	$5\ 10^4$	$1.6\ 10^5$	$6.9\ 10^{12}$
^{241}Am	$1.5\ 10^{-3}$	10	10^4	$2.2\ 10^5$	$3.0\ 10^{12}$
^{243}Am	$8.8\ 10^{-5}$	10	10^4	$2.2\ 10^5$	$3.0\ 10^{12}$

Notes:

(1) f_f is the concentration factor for fish

(2) K is the distribution coefficient between sediments and water

(3) F_i is the collective effective dose equivalent per unit activity ingested collectively

(4) F_a is the effective dose equivalent per unit infinite-time integral of the breathing air concentration.

parameters of Table 4 and the annual potential fish catch from the ocean. The implicit assumption, which makes it unnecessary to know average fish intakes, is that all fish caught is eventually eaten. The annual potential fish catch is taken to be of the order of $5\ 10^{7}$ tonnes (68).

68. Near costal areas the resuspension of contaminated sediments would originate an inhalation pathway. To assess this contribution, in addition to the pertinent values of Table 4, it has been assumed that the depth of resuspendable sediments is 0.1 cm and that the resuspension factor is 10^{-9} cm^{-1} (69). It was also assumed that a total of 10^{8} individuals will be exposed to this route (about 1 per cent of the assumed future population size). Fractional time for outdoors exposure has been taken to be 20 per cent.

69. Using the parameters of Table 4 and a conservative model for external exposures to sediments (semi-infinite geometry), it can be shown readily that the contribution of these exposures to the collective dose commitment is negligible compared to that of the two pathways discussed previously.

70. Unconventional pathways may be added in the future, affecting the collective dose commitment estimates. Among these, drinking water from desalted sea water and consumption of other marine products are worth mentioning. Desalted water contribution to the collective dose commitment appears to be negligible compared to the pathways included in Tables 5, even if it is conservatively assumed that 10^{8} individuals derive their drinking water from this source. Desalination processes will separate most nuclides from the water, with varying decontamination factors, probably in the order of 10^{2}.

71. Harvesting from the sea might be further developed in the future. Small zooplankton seems a likely candidate because its harvesting may increase the available proteins by an order of magnitude. As the concentration factors for many radionuclides in plankton is substantially higher than in fish, the contribution to the collective dose commitment might conceptually be significant. However, such a development seems unlikely, even in the long term, because the harvesting and processing of small zooplankton is likely to consume disproportionate amounts of energy (70).

72. The estimated annual potential of red crab is in the order of 10^6 tonnes. The possibility for developing a fishery is considered, but the lower annual potential as compared to fish, makes the possible collective dose commitment contribution not exceeding that of fish, even assuming larger concentration factors. The fishery of cephalopods could eventually be developed, the estimated annual potential being of the order of 10^7 tonnes. Concentration factors for cephalopods are not well known, but could be higher than those for fish for some radionuclides. Taking into account the relative composition of the waste as a function of age, it seems possible that the collective dose commitment for the younger ages (10^4 - 10^5 years) might be increased by the postulated cephalopode pathway (by an order of magnitude), while that for the older ages (10^5 - 10^6 years) is practically not affected.

4.3.2. Estimates of Collective Dose Commitment

73. The collective dose commitment for a given input of radionuclides is independent of input rate and duration. On the assumption that the entire disposed mixture migrates eventually to circulating waters, to arrive

there only much "older" (paragraph 50), the information presented in the preceding paragraphs allows the calculation of the collective dose commitment per GW(e)y as a function of the age of the waste mixture when entering circulating water bodies. Tables 5.1, 5.2, 5.3, 5.4, 5.5, 5.6 and 5.7, summarize the results of these calculations for the seven fuel cycle strategies under consideration.

74. According to the time delays estimated for the isolation provided by engineered and geological factors, the relevant range of collective dose commitments can be obtained from the columns 10^5 years and 10^6 years of these Tables. The range of values are:

	man rem / GW(e)y
Fuel cycle strategy 1	$8.9\ 10^3$ - $2.7\ 10^4$
Fuel cycle strategy 2	$9.7\ 10^2$ - $2.8\ 10^3$
Fuel cycle strategy 3	$1.8\ 10^3$ - $5.2\ 10^3$
Fuel cycle strategy 4	$3.1\ 10^4$ - $4.5\ 10^4$
Fuel cycle strategy 5	$6.8\ 10^2$ - $4.6\ 10^3$
Fuel cycle strategy 6	$3.6\ 10^3$ - $2.6\ 10^4$
Fuel cycle strategy 7	$2.5\ 10^3$ - $1.2\ 10^4$

75. The values given above show that waste management of high level wastes or of spent fuel results in the same order of magnitude of collective dose commitment than the reported total contributions of operations of the fuel cycle other than waste management (1). The actual delivery of doses comprised in the commitment from waste repositories spreads much more in the future than those from these other operations, and as a result the future annual individual doses will be much smaller in the case of waste repositories.

76. For comparison, collective dose commitments have also been calculated for the input assessed for a reference salt

Table 5.1

Normalized collective dose commitment as a function of the time of arrival of the radionuclides to circulating waters
Fuel Cycle I: Light Water Reactor with spent fuel disposal

Nuclide	Age of waste mixture							
	10^3 y		10^4 y		10^5 y		10^6 y	
	(1)	(2)	(1)	(2)	(1)	(2)	(1)	(2)
	Normalized collective dose commitment (man rem per Ci/y)							
^{99}Tc	$6.0\ 10^1$	$9.5\ 10^0$	$5.8\ 10^1$	$9.1\ 10^0$	$4.3\ 10^1$	$6.8\ 10^0$	$2.2\ 10^0$	$3.4\ 10^{-1}$
^{129}I	$9.8\ 10^1$	$2.1\ 10^1$	$9.8\ 10^1$	$2.1\ 10^1$	$9.8\ 10^1$	$2.1\ 10^1$	$9.0\ 10^1$	$1.9\ 10^1$
^{135}Cs	$7.5\ 10^0$	$3.1\ 10^2$	$7.5\ 10^0$	$3.1\ 10^2$	$7.3\ 10^0$	$3.0\ 10^2$	$5.9\ 10^0$	$2.5\ 10^2$
^{226}Ra	$3.2\ 10^1$	$7.6\ 10^0$	$1.4\ 10^3$	$3.4\ 10^2$	$1.1\ 10^4$	$2.8\ 10^3$	$5.7\ 10^3$	$1.4\ 10^3$
^{234}U	$1.9\ 10^2$	$1.1\ 10^3$	$1.9\ 10^2$	$1.0\ 10^3$	$1.6\ 10^2$	$8.6\ 10^2$	$4.6\ 10^1$	$2.5\ 10^2$
^{235}U	$1.8\ 10^0$	$1.3\ 10^1$	$2.1\ 10^0$	$1.5\ 10^1$	$3.0\ 10^0$	$2.1\ 10^0$	$3.1\ 10^0$	$2.1\ 10^0$
^{236}U	$3.3\ 10^1$	$4.3\ 10^1$	$4.3\ 10^1$	$5.6\ 10^1$	$5.0\ 10^1$	$6.4\ 10^1$	$5.0\ 10^1$	$6.4\ 10^1$
^{238}U	$3.6\ 10^1$	$2.5\ 10^2$	$3.6\ 10^1$	$2.5\ 10^2$	$3.6\ 10^1$	$2.5\ 10^2$	$3.6\ 10^1$	$2.5\ 10^2$
^{237}Np	$2.0\ 10^2$	$5.6\ 10^2$	$2.3\ 10^2$	$6.5\ 10^2$	$2.2\ 10^2$	$6.3\ 10^2$	$1.6\ 10^2$	$4.6\ 10^2$
^{238}Pu	$2.5\ 10^2$	$2.6\ 10^1$	-	-	-	-	-	-
^{239}Pu	$5.3\ 10^4$	$1.2\ 10^5$	$4.0\ 10^4$	$9.1\ 10^4$	$3.0\ 10^3$	$6.9\ 10^3$	-	-
^{240}Pu	$7.2\ 10^4$	$1.1\ 10^5$	$2.9\ 10^4$	$4.3\ 10^4$	$2.9\ 10^0$	$4.3\ 10^0$	-	-
^{241}Pu	$4.3\ 10^1$	$9.4\ 10^{-2}$	$2.1\ 10^1$	$4.5\ 10^{-2}$	$1.1\ 10^{-2}$	-	-	-
^{242}Pu	$2.4\ 10^2$	$6.5\ 10^2$	$2.4\ 10^2$	$6.4\ 10^2$	$2.0\ 10^2$	$5.5\ 10^2$	$3.8\ 10^1$	$1.0\ 10^2$
^{241}Am	$1.7\ 10^5$	$6.0\ 10^3$	$4.1\ 10^1$	$1.4\ 10^0$	$2.0\ 10^{-2}$	-	-	-
^{243}Am	$3.9\ 10^3$	$3.6\ 10^3$	$1.7\ 10^3$	$1.6\ 10^3$	$5.3\ 10^{-1}$	$4.9\ 10^{-1}$	-	-
Sub-totals	$3.0\ 10^5$	$2.4\ 10^5$	$7.3\ 10^4$	$1.4\ 10^5$	$1.5\ 10^4$	$1.2\ 10^4$	$6.1\ 10^3$	$2.8\ 10^3$
Totals	$5.4\ 10^5$		$2.1\ 10^5$		$2.7\ 10^4$		$8.9\ 10^3$	

(1) Fresh water collective dose contribution. (2) Ocean collective dose contribution.

Table 5.2

Normalized collective dose commitment as a function of the time of arrival of the radionuclides to circulating waters
Fuel Cycle 2: Light Water Reactor with plutonium recycle

Nuclide	Age of waste mixture							
	10^3 y		10^4 y		10^5 y		10^6 y	
	(1)	(2)	(1)	(2)	(1)	(2)	(1)	(2)
	Normalized collective dose commitment (man rem per GW(e)y)							
^{99}Tc	$6.0\ 10^{1}$	$9.5\ 10^{0}$	$5.8\ 10^{1}$	$9.1\ 10^{0}$	$4.3\ 10^{1}$	$6.8\ 10^{0}$	$2.2\ 10^{0}$	$3.4\ 10^{-1}$
^{129}I	$9.8\ 10^{1}$	$2.1\ 10^{1}$	$9.8\ 10^{1}$	$2.1\ 10^{1}$	$9.8\ 10^{1}$	$2.1\ 10^{1}$	$9.0\ 10^{1}$	$1.9\ 10^{1}$
^{135}Cs	$1.1\ 10^{1}$	$4.6\ 10^{2}$	$1.1\ 10^{1}$	$4.6\ 10^{2}$	$1.1\ 10^{1}$	$4.6\ 10^{2}$	$8.6\ 10^{0}$	$3.6\ 10^{2}$
^{226}Ra	$7.2\ 10^{-1}$	$1.7\ 10^{-1}$	$3.8\ 10^{1}$	$9.2\ 10^{0}$	$3.2\ 10^{2}$	$7.7\ 10^{1}$	$9.9\ 10^{1}$	$2.4\ 10^{1}$
^{234}U	$5.6\ 10^{0}$	$3.1\ 10^{1}$	$5.6\ 10^{0}$	$3.1\ 10^{1}$	$4.3\ 10^{0}$	$2.3\ 10^{1}$	$7.3\ 10^{-1}$	$4.0\ 10^{0}$
^{235}U	$2.0\ 10^{-2}$	$1.4\ 10^{-1}$	$3.3\ 10^{-2}$	$2.3\ 10^{-1}$	$1.3\ 10^{-1}$	$9.2\ 10^{-1}$	$1.4\ 10^{-1}$	$9.7\ 10^{-1}$
^{236}U	$5.0\ 10^{-1}$	$6.4\ 10^{-1}$	$1.4\ 10^{0}$	$1.8\ 10^{0}$	$1.9\ 10^{0}$	$2.5\ 10^{0}$	$1.9\ 10^{0}$	$2.4\ 10^{0}$
^{238}U	$4.0\ 10^{-1}$	$2.8\ 10^{0}$	$4.0\ 10^{-1}$	$2.8\ 10^{0}$	$4.0\ 10^{-1}$	$2.8\ 10^{0}$	$4.0\ 10^{-1}$	$2.8\ 10^{0}$
^{237}Np	$1.1\ 10^{2}$	$3.1\ 10^{2}$	$1.1\ 10^{2}$	$3.2\ 10^{2}$	$1.1\ 10^{2}$	$3.1\ 10^{2}$	$8.4\ 10^{1}$	$2.4\ 10^{2}$
^{238}Pu	$1.2\ 10^{2}$	$1.2\ 10^{1}$	-	-	-	-	-	-
^{239}Pu	$1.2\ 10^{3}$	$2.6\ 10^{3}$	$2.8\ 10^{3}$	$6.5\ 10^{3}$	$4.0\ 10^{2}$	$9.2\ 10^{2}$	-	-
^{240}Pu	$7.7\ 10^{3}$	$1.2\ 10^{4}$	$3.1\ 10^{3}$	$4.7\ 10^{3}$	$3.5\ 10^{-1}$	$5.3\ 10^{-1}$	-	-
^{241}Pu	$4.6\ 10^{2}$	$1.0\ 10^{0}$	$2.2\ 10^{2}$	$4.9\ 10^{-1}$	$1.2\ 10^{-1}$	-	-	-
^{242}Pu	$5.3\ 10^{0}$	$1.4\ 10^{1}$	$5.3\ 10^{0}$	$1.4\ 10^{1}$	$4.4\ 10^{0}$	$1.2\ 10^{1}$	$8.6\ 10^{-1}$	$2.3\ 10^{0}$
^{241}Am	$2.5\ 10^{4}$	$8.7\ 10^{2}$	$4.4\ 10^{2}$	$1.6\ 10^{1}$	$2.5\ 10^{-1}$	$9.0\ 10^{-3}$	-	-
^{243}Am	$1.7\ 10^{4}$	$1.6\ 10^{4}$	$7.3\ 10^{3}$	$6.7\ 10^{3}$	$2.3\ 10^{0}$	$2.1\ 10^{0}$	-	-
Sub-totals	$5.2\ 10^{4}$	$3.2\ 10^{4}$	$1.4\ 10^{4}$	$1.9\ 10^{4}$	$1.0\ 10^{3}$	$1.8\ 10^{3}$	$2.9\ 10^{2}$	$6.8\ 10^{2}$
Totals	$8.4\ 10^{4}$		$3.3\ 10^{4}$		$2.8\ 10^{3}$		$9.7\ 10^{2}$	

(1) [illegible] collective dose contribution. (2) Ocean collective dose contribution

Table 5.3

Normalized collective dose commitment as a function of the time of arrival of the radionuclides to circulating waters
Fuel Cycle 3: Fast Breeder Reactor with plutonium recycle

Nuclide	Age of waste mixture							
	10^3 y		10^4 y		10^5 y		10^6 y	
	(1)	(2)	(1)	(2)	(1)	(2)	(1)	(2)
	Normalized collective dose commitment (man rem per GW(e)y)							
^{99}Tc	$5.4\ 10^{1}$	$8.6\ 10^{0}$	$5.3\ 10^{1}$	$8.4\ 10^{0}$	$3.8\ 10^{1}$	$6.1\ 10^{0}$	$2.0\ 10^{0}$	$3.2\ 10^{-1}$
^{129}I	$6.3\ 10^{1}$	$1.3\ 10^{1}$	$6.3\ 10^{1}$	$1.3\ 10^{1}$	$6.3\ 10^{1}$	$1.3\ 10^{1}$	$6.1\ 10^{1}$	$1.3\ 10^{1}$
^{135}Cs	$3.1\ 10^{1}$	$1.3\ 10^{3}$	$3.1\ 10^{1}$	$1.3\ 10^{3}$	$3.0\ 10^{1}$	$1.3\ 10^{3}$	$2.5\ 10^{1}$	$1.0\ 10^{3}$
^{226}Ra	$8.4\ 10^{-1}$	$2.0\ 10^{-1}$	$1.4\ 10^{2}$	$3.3\ 10^{1}$	$1.1\ 10^{3}$	$2.6\ 10^{2}$	$2.3\ 10^{2}$	$5.5\ 10^{1}$
^{234}U	$1.9\ 10^{1}$	$1.0\ 10^{2}$	$1.9\ 10^{1}$	$1.0\ 10^{2}$	$1.5\ 10^{1}$	$8.1\ 10^{1}$	$1.4\ 10^{0}$	$7.7\ 10^{0}$
^{235}U	$6.6\ 10^{-3}$	$4.6\ 10^{-2}$	$4.0\ 10^{-2}$	$2.8\ 10^{-1}$	$1.9\ 10^{-1}$	$1.3\ 10^{0}$	$2.0\ 10^{-1}$	$1.4\ 10^{0}$
^{236}U	$1.8\ 10^{-1}$	$2.3\ 10^{-1}$	$1.1\ 10^{0}$	$1.5\ 10^{0}$	$1.8\ 10^{0}$	$2.3\ 10^{0}$	$1.7\ 10^{0}$	$2.3\ 10^{0}$
^{238}U	$2.3\ 10^{-1}$	$1.6\ 10^{0}$	$2.3\ 10^{-1}$	$1.6\ 10^{0}$	$2.3\ 10^{-1}$	$1.6\ 10^{0}$	$2.3\ 10^{-1}$	$1.6\ 10^{0}$
^{237}Np	$1.0\ 10^{2}$	$2.9\ 10^{2}$	$1.3\ 10^{2}$	$3.6\ 10^{2}$	$1.2\ 10^{2}$	$3.4\ 10^{2}$	$9.0\ 10^{1}$	$2.6\ 10^{2}$
^{238}Pu	$3.4\ 10^{2}$	$3.4\ 10^{1}$	-	-	-	-	-	-
^{239}Pu	$6.2\ 10^{3}$	$1.4\ 10^{4}$	$5.8\ 10^{3}$	$1.3\ 10^{4}$	$5.3\ 10^{2}$	$1.2\ 10^{3}$	-	-
^{240}Pu	$7.7\ 10^{3}$	$1.2\ 10^{4}$	$3.3\ 10^{3}$	$4.9\ 10^{3}$	$3.4\ 10^{-1}$	$5.1\ 10^{-1}$	-	-
^{241}Pu	$3.6\ 10^{3}$	$8.0\ 10^{0}$	$1.7\ 10^{3}$	$3.7\ 10^{0}$	-	-	-	-
^{242}Pu	$2.6\ 10^{1}$	$7.2\ 10^{1}$	$2.6\ 10^{1}$	$7.2\ 10^{1}$	$2.2\ 10^{1}$	$6.0\ 10^{1}$	$4.3\ 10^{0}$	$1.2\ 10^{1}$
^{241}Am	$2.0\ 10^{5}$	$7.1\ 10^{3}$	$6.2\ 10^{-1}$	$2.2\ 10^{-2}$	-	-	-	-
^{243}Am	$9.2\ 10^{3}$	$8.5\ 10^{3}$	$4.0\ 10^{3}$	$3.7\ 10^{3}$	$1.3\ 10^{0}$	$1.2\ 10^{0}$	-	-
Sub-totals	$2.3\ 10^{5}$	$4.4\ 10^{4}$	$1.5\ 10^{4}$	$2.4\ 10^{4}$	$1.9\ 10^{3}$	$3.3\ 10^{3}$	$4.2\ 10^{2}$	$1.4\ 10^{3}$
Totals	$2.7\ 10^{5}$		$3.9\ 10^{4}$		$5.2\ 10^{3}$		$1.8\ 10^{3}$	

(1) Fresh water collective dose contribution. (2) Ocean collective dose contribution.

Table 5.-

Normalized collective dose commitment as a function of the time of arrival of the radionuclides to circulating waters

Fuel Cycle 4: Heavy Water Reactor with spent fuel disposal

Nuclide	Age of food ingestion							
	10^3 y		10^4 y		10^5 y		10^6 y	
	(1)	(2)	(1)	(2)	(1)	(2)	(1)	(2)
	Normalized collective dose commitment (man rem per GW(e)y)							
^{99}Tc	$2.5\ 10^{1}$	$4.0\ 10^{0}$	$2.4\ 10^{1}$	$3.8\ 10^{0}$	$1.8\ 10^{1}$	$2.8\ 10^{0}$	$9.2\ 10^{-1}$	$1.5\ 10^{-1}$
^{129}I	$9.0\ 10^{1}$	$1.9\ 10^{1}$	$9.0\ 10^{1}$	$1.9\ 10^{1}$	$9.0\ 10^{1}$	$1.9\ 10^{1}$	$9.0\ 10^{1}$	$1.9\ 10^{1}$
^{135}Cs	$3.5\ 10^{0}$	$1.4\ 10^{2}$	$3.5\ 10^{0}$	$1.4\ 10^{2}$	$3.4\ 10^{0}$	$1.4\ 10^{2}$	$2.7\ 10^{0}$	$1.1\ 10^{2}$
^{226}Ra	$3.2\ 10^{1}$	$7.8\ 10^{0}$	$1.3\ 10^{3}$	$3.1\ 10^{2}$	$1.2\ 10^{4}$	$2.9\ 10^{3}$	$2.2\ 10^{4}$	$5.3\ 10^{3}$
^{234}U	$1.8\ 10^{2}$	$9.7\ 10^{2}$	$1.8\ 10^{2}$	$9.7\ 10^{2}$	$1.8\ 10^{2}$	$9.9\ 10^{2}$	$1.9\ 10^{2}$	$1.1\ 10^{3}$
^{235}U	$3.2\ 10^{0}$	$2.2\ 10^{1}$	$4.0\ 10^{0}$	$2.8\ 10^{1}$	$5.9\ 10^{0}$	$4.1\ 10^{1}$	$6.3\ 10^{0}$	$4.4\ 10^{1}$
^{236}U	$3.0\ 10^{1}$	$4.0\ 10^{1}$	$5.0\ 10^{1}$	$6.4\ 10^{1}$	$6.3\ 10^{1}$	$8.2\ 10^{1}$	$5.9\ 10^{1}$	$7.7\ 10^{1}$
^{238}U	$1.9\ 10^{2}$	$1.4\ 10^{3}$	$1.9\ 10^{2}$	$1.4\ 10^{3}$	$1.9\ 10^{2}$	$1.4\ 10^{3}$	$1.9\ 10^{2}$	$1.4\ 10^{3}$
^{237}Np	$1.2\ 10^{2}$	$3.4\ 10^{2}$	$1.4\ 10^{2}$	$4.1\ 10^{2}$	$1.4\ 10^{2}$	$4.1\ 10^{2}$	$1.1\ 10^{2}$	$3.1\ 10^{2}$
^{238}Pu	$1.9\ 10^{2}$	$2.0\ 10^{1}$	-	-	-	-	-	-
^{239}Pu	$1.3\ 10^{5}$	$3.1\ 10^{5}$	$1.1\ 10^{5}$	$2.4\ 10^{5}$	$7.7\ 10^{3}$	$1.8\ 10^{4}$	-	-
^{240}Pu	$1.6\ 10^{5}$	$2.2\ 10^{5}$	$6.7\ 10^{4}$	$1.0\ 10^{5}$	$6.7\ 10^{0}$	$1.0\ 10^{1}$	-	-
^{241}Pu	$1.2\ 10^{-1}$	-	$5.6\ 10^{-2}$	-	-	-	-	-
^{242}Pu	$1.5\ 10^{2}$	$4.0\ 10^{2}$	$1.4\ 10^{2}$	$3.9\ 10^{2}$	$1.2\ 10^{2}$	$3.4\ 10^{2}$	$2.4\ 10^{1}$	$6.5\ 10^{1}$
^{241}Am	$1.4\ 10^{5}$	$4.8\ 10^{3}$	$7.2\ 10^{-2}$	$2.5\ 10^{-3}$	-	-	-	-
^{243}Am	$3.8\ 10^{2}$	$3.5\ 10^{2}$	$1.7\ 10^{2}$	$1.6\ 10^{2}$	$5.2\ 10^{-2}$	$4.8\ 10^{-2}$	-	-
Sub-totals	$4.3\ 10^{5}$	$5.4\ 10^{5}$	$1.8\ 10^{5}$	$3.4\ 10^{5}$	$2.1\ 10^{4}$	$2.4\ 10^{4}$	$2.3\ 10^{3}$	$8.4\ 10^{3}$
Totals	$9.7\ 10^{5}$		$5.2\ 10^{5}$		$4.5\ 10^{4}$		$3.1\ 10^{4}$	

(1) Through water collective dose contributions. (2) Ingestion collective dose contributions.

Table 5.5

Normalized collective dose commitment as a function of the time of arrival of the radionuclides to circulating waters
Fuel Cycle 5: Heavy water Reactor with plutonium recycle

Nuclide	Age of waste mixture							
	10^3 y		10^4 y		10^5 y		10^6 y	
	(1)	(2)	(1)	(2)	(1)	(2)	(1)	(2)
	Normalized collective dose commitment (man rem per GW(e)y)							
^{99}Tc	$6.5\ 10^{1}$	$1.0\ 10^{1}$	$6.2\ 10^{1}$	$9.9\ 10^{0}$	$4.7\ 10^{1}$	$7.4\ 10^{0}$	$2.4\ 10^{0}$	$3.8\ 10^{-1}$
^{129}I	-	-	-	-	-	-	-	-
^{135}Cs	$4.6\ 10^{0}$	$1.9\ 10^{2}$	$4.6\ 10^{0}$	$1.9\ 10^{2}$	$4.5\ 10^{0}$	$1.9\ 10^{2}$	$3.6\ 10^{0}$	$1.5\ 10^{2}$
^{226}Ra	-	-	$9.9\ 10^{0}$	$2.4\ 10^{0}$	$8.4\ 10^{1}$	$2.0\ 10^{1}$	$9.9\ 10^{1}$	$2.4\ 10^{1}$
^{234}U	$1.4\ 10^{0}$	$7.6\ 10^{0}$	$1.4\ 10^{0}$	$7.6\ 10^{0}$	$1.3\ 10^{0}$	$6.8\ 10^{0}$	$8.6\ 10^{-1}$	$4.7\ 10^{0}$
^{235}U	$7.9\ 10^{-3}$	$5.5\ 10^{-2}$	$4.0\ 10^{-2}$	$2.8\ 10^{-1}$	$3.0\ 10^{-1}$	$2.1\ 10^{0}$	$3.3\ 10^{-1}$	$2.3\ 10^{0}$
^{236}U	$3.1\ 10^{-1}$	$4.0\ 10^{-1}$	$1.8\ 10^{0}$	$2.3\ 10^{0}$	$2.8\ 10^{0}$	$3.6\ 10^{0}$	$2.7\ 10^{0}$	$3.5\ 10^{0}$
^{238}U	$8.2\ 10^{-1}$	$5.8\ 10^{0}$	$8.2\ 10^{-1}$	$5.8\ 10^{0}$	$8.2\ 10^{-1}$	$5.8\ 10^{0}$	$8.2\ 10^{-1}$	$5.8\ 10^{0}$
^{237}Np	$1.0\ 10^{2}$	$2.9\ 10^{2}$	$1.2\ 10^{2}$	$3.4\ 10^{2}$	$1.2\ 10^{2}$	$3.4\ 10^{2}$	$9.0\ 10^{1}$	$2.6\ 10^{2}$
^{238}Pu	$1.2\ 10^{1}$	$1.3\ 10^{0}$	-	-	-	-	-	-
^{239}Pu	$1,8\ 10^{3}$	$4.1\ 10^{3}$	$7.2\ 10^{3}$	$1.6\ 10^{4}$	$1.1\ 10^{3}$	$2.5\ 10^{3}$	-	-
^{240}Pu	$1.2\ 10^{4}$	$1.9\ 10^{4}$	$5.3\ 10^{3}$	$7.9\ 10^{3}$	$5.3\ 10^{-1}$	$7.9\ 10^{-1}$	-	-
^{241}Pu	$2.7\ 10^{1}$	$5.9\ 10^{-2}$	$1.2\ 10^{1}$	$2.6\ 10^{-2}$	$7.3\ 10^{-3}$	-	-	-
^{242}Pu	$3.5\ 10^{1}$	$9.5\ 10^{1}$	$3.5\ 10^{1}$	$9.4\ 10^{1}$	$2.9\ 10^{1}$	$7.9\ 10^{1}$	$5.8\ 10^{0}$	$1.6\ 10^{1}$
^{241}Am	$1.0\ 10^{5}$	$3.7\ 10^{3}$	$6.0\ 10^{-2}$	$2.1\ 10^{-3}$	-	-	-	-
^{243}Am	$5.2\ 10^{4}$	$4.8\ 10^{4}$	$2.3\ 10^{4}$	$2.1\ 10^{4}$	$7.3\ 10^{0}$	$6.7\ 10^{0}$	-	-
Sub-totals	$1.7\ 10^{5}$	$7.5\ 10^{4}$	$3.6\ 10^{4}$	$4.6\ 10^{4}$	$1.4\ 10^{3}$	$3.2\ 10^{3}$	$2.1\ 10^{2}$	$4.7\ 10^{2}$
Totals	$2.4\ 10^{5}$		$8.2\ 10^{4}$		$4.6\ 10^{3}$		$6.8\ 10^{2}$	

(1) Fresh water collective dose contribution. (2) Ocean collective dose contribution.

Table 5.6

Normalized collective dose co-mitment as a function of the time of arrival of the radionuclides to circulating waters

Fuel Cycle 6: Heavy Water Reactor with uranium - thorium recycle

Nuclide	Age of waste mixture							
	10^3 y		10^4 y		10^5 y		10^6 y	
	(1)	(2)	(1)	(2)	(1)	(2)	(1)	(2)
	Normalized collective dose commitment (man rem per GW(e)y)							
^{99}Tc	$2.0\ 10^1$	$3.2\ 10^0$	$1.9\ 10^1$	$3.0\ 10^0$	$1.4\ 10^1$	$2.3\ 10^0$	$7.6\ 10^{-1}$	$1.2\ 10^{-1}$
^{129}I	-	-	-	-	-	-	-	-
^{135}Cs	$5.6\ 10^0$	$2.4\ 10^2$	$5.6\ 10^0$	$2.4\ 10^2$	$5.6\ 10^0$	$2.3\ 10^2$	$4.5\ 10^0$	$1.9\ 10^2$
^{226}Ra	$3.2\ 10^1$	$7.8\ 10^0$	$1.3\ 10^3$	$3.1\ 10^2$	$1.0\ 10^4$	$2.5\ 10^3$	$1.9\ 10^3$	$4.6\ 10^2$
^{228}Th	$1.4\ 10^0$	$2.2\ 10^0$	$1.4\ 10^0$	$2.2\ 10^0$	$1.4\ 10^0$	$2.2\ 10^0$	$1.4\ 10^0$	$2.2\ 10^0$
^{229}Th	$1.8\ 10^2$	$2.4\ 10^2$	$1.2\ 10^3$	$1.6\ 10^3$	$1.4\ 10^3$	$2.0\ 10^3$	$9.4\ 10^1$	$1.3\ 10^2$
^{230}Th	$5.1\ 10^0$	$7.8\ 10^0$	$4.8\ 10^1$	$7.5\ 10^1$	$3.0\ 10^2$	$4.6\ 10^2$	$5.5\ 10^1$	$8.5\ 10^1$
^{232}Th	$1.4\ 10^0$	$2.2\ 10^0$	$1.4\ 10^0$	$2.2\ 10^0$	$1.4\ 10^0$	$2.2\ 10^0$	$1.4\ 10^0$	$2.2\ 10^0$
^{232}U	-	-	-	-	-	-	-	-
^{233}U	$5.9\ 10^2$	$9.0\ 10^3$	$5.6\ 10^2$	$8.5\ 10^3$	$4.3\ 10^2$	$6.5\ 10^3$	$2.8\ 10^1$	$4.2\ 10^2$
^{234}U	$1.8\ 10^2$	$9.7\ 10^2$	$1.7\ 10^2$	$9.5\ 10^2$	$1.4\ 10^2$	$7.4\ 10^2$	$1.1\ 10^1$	$6.1\ 10^1$
^{235}U	$1.2\ 10^{-2}$	$8.3\ 10^{-2}$	$2.4\ 10^{-2}$	$1.7\ 10^{-1}$	$6.3\ 10^{-2}$	$4.4\ 10^{-1}$	$6.6\ 10^{-2}$	$4.6\ 10^{-1}$
^{236}U	$8.2\ 10^{-1}$	$1.1\ 10^0$	$1.0\ 10^0$	$1.4\ 10^0$	$1.2\ 10^0$	$1.6\ 10^0$	$1.2\ 10^0$	$1.5\ 10^0$
^{238}U	$7.9\ 10^{-2}$	$5.5\ 10^{-2}$	$7.9\ 10^{-2}$	$5.5\ 10^{-2}$	$7.9\ 10^{-2}$	$5.5\ 10^{-2}$	$7.9\ 10^{-2}$	$5.5\ 10^{-2}$
^{237}Np	$5.4\ 10^1$	$1.5\ 10^2$	$5.5\ 10^1$	$1.4\ 10^3$	$5.3\ 10^1$	$1.5\ 10^2$	$4.0\ 10^1$	$1.1\ 10^2$
^{238}Pu	$4.3\ 10^1$	$4.4\ 10^0$	-	-	-	-	-	-
^{239}Pu	$2.3\ 10^3$	$5.3\ 10^3$	$1.8\ 10^3$	$4.1\ 10^3$	$1.3\ 10^2$	$3.0\ 10^2$	-	-
^{240}Pu	$2.0\ 10^3$	$3.0\ 10^3$	$8.2\ 10^2$	$1.2\ 10^3$	$8.2\ 10^{-2}$	$1.2\ 10^{-1}$	-	-
^{241}Pu	$2.9\ 10^1$	$6.3\ 10^{-2}$	$1.4\ 10^1$	$3.0\ 10^{-2}$	$7.3\ 10^{-3}$	-	-	-
^{242}Pu	$2.5\ 10^0$	$6.8\ 10^0$	$2.4\ 10^0$	$6.6\ 10^0$	$2.1\ 10^0$	$5.6\ 10^0$	$4.0\ 10^{-1}$	$1.1\ 10^0$
^{241}Am	$3.5\ 10^3$	$1.2\ 10^2$	$1.8\ 10^{-3}$	-	-	-	-	-
^{243}Am	$6.6\ 10^0$	$6.1\ 10^0$	$3.0\ 10^0$	$2.7\ 10^0$	-	-	-	-
Sub-totals	$8.9\ 10^3$	$1.9\ 10^4$	$6.0\ 10^3$	$1.8\ 10^4$	$1.3\ 10^4$	$1.3\ 10^4$	$2.1\ 10^3$	$1.5\ 10^3$
Totals	$2.8\ 10^4$		$2.4\ 10^4$		$2.6\ 10^4$		$3.6\ 10^3$	

(1) Fresh water collective dose contribution. (2) Ocean collective dose contribution.

Table 5.7

Normalized collective dose commitment as a function of the time of arrival of the radionuclides to circulating waters
Fuel Cycle 7: High Temperature Reactor with uranium - thorium recycle

Nuclide	Age of waste mixture							
	10^3 y		10^4 y		10^5 y		10^6 y	
	(1)	(2)	(1)	(2)	(1)	(2)	(1)	(2)
	Normalized collective dose commitment (man rem per GW(e)y)							
^{99}Tc	$5.9\ 10^{0}$	$5.3\ 10^{-1}$	$5.6\ 10^{0}$	$8.9\ 10^{-1}$	$4.2\ 10^{0}$	$6.6\ 10^{-1}$	$2.2\ 10^{-1}$	$3.4\ 10^{-2}$
^{129}I	$5.2\ 10^{1}$	$1.1\ 10^{1}$	$5.2\ 10^{1}$	$1.1\ 10^{1}$	$5.2\ 10^{1}$	$1.1\ 10^{1}$	$5.0\ 10^{1}$	$1.1\ 10^{1}$
^{135}Cs	$4.3\ 10^{0}$	$1.8\ 10^{2}$	$4.3\ 10^{0}$	$1.8\ 10^{2}$	$4.2\ 10^{0}$	$1.7\ 10^{2}$	$3.4\ 10^{0}$	$1.4\ 10^{2}$
^{226}Ra	$3.0\ 10^{1}$	$7.4\ 10^{0}$	$8.4\ 10^{2}$	$2.0\ 10^{2}$	$6.5\ 10^{3}$	$1.6\ 10^{3}$	$1.2\ 10^{3}$	$2.8\ 10^{2}$
^{228}Th	$2.0\ 10^{-1}$	$3.1\ 10^{-1}$	$1.7\ 10^{-1}$	$2.6\ 10^{-1}$	$1.7\ 10^{-1}$	$2.6\ 10^{-1}$	$1.7\ 10^{-1}$	$2.6\ 10^{-1}$
^{229}Th	$3.6\ 10^{1}$	$5.0\ 10^{1}$	$2.4\ 10^{2}$	$3.3\ 10^{2}$	$3.0\ 10^{2}$	$4.1\ 10^{2}$	$7.9\ 10^{1}$	$1.1\ 10^{2}$
^{230}Th	$4.0\ 10^{0}$	$6.1\ 10^{0}$	$3.1\ 10^{1}$	$4.8\ 10^{1}$	$1.9\ 10^{2}$	$2.9\ 10^{2}$	$3.4\ 10^{1}$	$5.3\ 10^{1}$
^{232}Th	$1.7\ 10^{-1}$	$2.6\ 10^{-1}$	$1.7\ 10^{-1}$	$2.6\ 10^{-1}$	$1.7\ 10^{-1}$	$2.6\ 10^{-1}$	$1.7\ 10^{-1}$	$2.6\ 10^{-1}$
^{232}U	$1.0\ 10^{-2}$	-	-	-	-	-	-	-
^{233}U	$1.2\ 10^{2}$	$1.8\ 10^{3}$	$1.2\ 10^{2}$	$1.8\ 10^{3}$	$8.6\ 10^{1}$	$1.3\ 10^{3}$	$2.4\ 10^{1}$	$3.6\ 10^{2}$
^{234}U	$1.1\ 10^{2}$	$6.1\ 10^{2}$	$1.1\ 10^{2}$	$5.9\ 10^{2}$	$8.3\ 10^{1}$	$4.5\ 10^{2}$	$6.9\ 10^{0}$	$3.8\ 10^{1}$
^{235}U	$5.9\ 10^{-2}$	$4.1\ 10^{-1}$	$5.9\ 10^{-2}$	$4.1\ 10^{-1}$	$6.6\ 10^{-2}$	$4.6\ 10^{-1}$	$6.6\ 10^{-2}$	$4.6\ 10^{-1}$
^{236}U	$9.9\ 10^{0}$	$1.3\ 10^{1}$	$9.9\ 10^{0}$	$1.3\ 10^{1}$	$9.9\ 10^{0}$	$1.3\ 10^{1}$	$9.6\ 10^{0}$	$1.2\ 10^{1}$
^{238}U	$1.4\ 10^{-2}$	$9.9\ 10^{-2}$	$1.4\ 10^{-2}$	$9.9\ 10^{-2}$	$1.4\ 10^{-2}$	$9.9\ 10^{-2}$	$1.4\ 10^{-2}$	$9.9\ 10^{-2}$
^{237}Np	$5.2\ 10^{1}$	$1.5\ 10^{2}$	$5.3\ 10^{1}$	$1.5\ 10^{2}$	$5.1\ 10^{1}$	$1.4\ 10^{2}$	$3.8\ 10^{1}$	$1.1\ 10^{2}$
^{238}Pu	$1.3\ 10^{2}$	$1.4\ 10^{1}$	-	-	-	-	-	-
^{239}Pu	$2.6\ 10^{2}$	$6.1\ 10^{2}$	$2.2\ 10^{2}$	$5.1\ 10^{2}$	$1.8\ 10^{1}$	$4.1\ 10^{1}$	-	-
^{240}Pu	$4.5\ 10^{2}$	$6.7\ 10^{2}$	$1.8\ 10^{2}$	$2.7\ 10^{2}$	$1.9\ 10^{-2}$	$2.8\ 10^{-2}$	-	-
^{241}Pu	-	-	-	-	-	-	-	-
^{242}Pu	$7.7\ 10^{0}$	$2.1\ 10^{1}$	$7.7\ 10^{0}$	$2.1\ 10^{1}$	$6.2\ 10^{0}$	$1.7\ 10^{1}$	$1.2\ 10^{0}$	$3.4\ 10^{0}$
^{241}Am	$2.0\ 10^{3}$	$6.9\ 10^{1}$	-	-	-	-	-	-
^{243}Am	$1.5\ 10^{3}$	$1.3\ 10^{2}$	$6.6\ 10^{1}$	$6.1\ 10^{1}$	$2.0\ 10^{-2}$	$1.8\ 10^{-2}$	-	-
Sub-totals	$3.4\ 10^{3}$	$4.3\ 10^{3}$	$1.9\ 10^{3}$	$4.2\ 10^{3}$	$7.3\ 10^{3}$	$4.4\ 10^{3}$	$1.4\ 10^{3}$	$1.1\ 10^{3}$
Totals	$7.7\ 10^{3}$		$6.1\ 10^{3}$		$1.2\ 10^{4}$		$2.5\ 10^{3}$	

(1) Fresh water collective dose contribution. (2) Ocean collective dose contribution.

Table S.a

Normalized collective dose commitment
(Assessed from a Geological Transport Model)*
man rem per GW(e)y

Nuclide	Strategy													
	1		2		3		4		5		6		7	
	(1)	(2)	(1)	(2)	(1)	(2)	(1)	(2)	(1)	(2)	(1)	(2)	(1)	(2)
^{99}Tc	$4.1\ 10^{-2}$	$6.5\ 10^{-3}$	$4.1\ 10^{-2}$	$6.5\ 10^{-3}$	$3.7\ 10^{-2}$	$5.9\ 10^{-3}$	$4.4\ 10^{-2}$	$7.0\ 10^{-3}$	$4.4\ 10^{-2}$	$7.0\ 10^{-3}$	$3.6\ 10^{-2}$	$5.7\ 10^{-3}$	$2.6\ 10^{-3}$	$4.2\ 10^{-4}$
^{129}I	$9.8\ 10^{1}$	$2.1\ 10^{1}$	$9.8\ 10^{1}$	$2.1\ 10^{1}$	$6.2\ 10^{1}$	$1.3\ 10^{1}$	$9.0\ 10^{1}$	$1.9\ 10^{1}$	$9.7\ 10^{1}$	$2.1\ 10^{1}$	$1.4\ 10^{2}$	$3.0\ 10^{1}$	$9.0\ 10^{1}$	$1.9\ 10^{1}$
^{135}Cs	$5.1\ 10^{0}$	$2.1\ 10^{2}$	$7.6\ 10^{0}$	$3.2\ 10^{2}$	$2.1\ 10^{1}$	$8.7\ 10^{1}$	$2.4\ 10^{0}$	$9.9\ 10^{1}$	$3.1\ 10^{0}$	$1.3\ 10^{2}$	$3.9\ 10^{0}$	$1.6\ 10^{2}$	$2.9\ 10^{0}$	$1.2\ 10^{2}$
^{226}Ra	$5.3\ 10^{4}$	$1.3\ 10^{4}$	$3.5\ 10^{4}$	$8.4\ 10^{3}$	$9.5\ 10^{1}$	$2.3\ 10^{1}$	$4.9\ 10^{4}$	$1.2\ 10^{4}$	$2.1\ 10^{4}$	$5.2\ 10^{3}$	$2.0\ 10^{3}$	$4.8\ 10^{2}$	$1.6\ 10^{4}$	$3.8\ 10^{3}$
^{229}Th	$5.8\ 10^{-2}$	$8.0\ 10^{-2}$	$2.1\ 10^{-2}$	$2.8\ 10^{-2}$	$3.1\ 10^{-2}$	$4.2\ 10^{-2}$	$2.6\ 10^{-2}$	$3.6\ 10^{-2}$	$2.4\ 10^{-2}$	$3.3\ 10^{-2}$	$5.8\ 10^{-3}$	$8.0\ 10^{-3}$	$1.0\ 10^{-2}$	$1.4\ 10^{-2}$
^{230}Th	$3.2\ 10^{1}$	$4.9\ 10^{1}$	$2.0\ 10^{1}$	$3.1\ 10^{1}$	$5.2\ 10^{-2}$	$8.0\ 10^{-2}$	$2.9\ 10^{1}$	$4.4\ 10^{1}$	$1.2\ 10^{1}$	$1.9\ 10^{1}$	$1.0\ 10^{0}$	$1.6\ 10^{0}$	$8.7\ 10^{0}$	$1.3\ 10^{1}$
^{232}Th	$2.2\ 10^{-2}$	$3.7\ 10^{-2}$	$9.9\ 10^{-3}$	$1.5\ 10^{-2}$	$1.0\ 10^{-3}$	$1.6\ 10^{-3}$	$3.3\ 10^{-2}$	$5.1\ 10^{-2}$	$4\ 2\ 10^{-3}$	$6.5\ 10^{-3}$	$1.5\ 10^{0}$	$2.4\ 10^{0}$	$1.8\ 10^{-1}$	$2.7\ 10^{-1}$
^{233}U	$5.0\ 10^{-1}$	$7.5\ 10^{0}$	$2.4\ 10^{-1}$	$3.7\ 10^{0}$	$2.7\ 10^{-1}$	$4.1\ 10^{0}$	$3.1\ 10^{-2}$	$4.8\ 10^{0}$	$2.7\ 10^{-1}$	$4.0\ 10^{0}$	$1.2\ 10^{-1}$	$1.8\ 10^{0}$	$1.2\ 10^{-1}$	$1.8\ 10^{0}$
^{234}U	$2.2\ 10^{2}$	$1.2\ 10^{3}$	$1.4\ 10^{2}$	$7.4\ 10^{2}$	$3.6\ 10^{-1}$	$2.0\ 10^{0}$	$2.0\ 10^{2}$	$1.1\ 10^{3}$	$8.6\ 10^{1}$	$4.7\ 10^{2}$	$7.6\ 10^{0}$	$4.1\ 10^{1}$	$6.3\ 10^{1}$	$3.4\ 10^{2}$
^{235}U**	$5.6\ 10^{0}$	$3.9\ 10^{1}$	$1.7\ 10^{0}$	$1.2\ 10^{1}$	$2.1\ 10^{-1}$	$1.5\ 10^{0}$	$6.6\ 10^{0}$	$4.6\ 10^{1}$	$1.1\ 10^{0}$	$7.4\ 10^{0}$	$1.6\ 10^{-1}$	$1.1\ 10^{0}$	$8.6\ 10^{-1}$	$6.0\ 10^{0}$
^{236}U	$4.3\ 10^{1}$	$5.6\ 10^{1}$	$1.7\ 10^{1}$	$2.2\ 10^{1}$	$1.7\ 10^{0}$	$2.3\ 10^{0}$	$5.6\ 10^{1}$	$7.3\ 10^{1}$	$6.9\ 10^{0}$	$9.0\ 10^{0}$	$1.2\ 10^{0}$	$1.5\ 10^{0}$	$9.2\ 10^{0}$	$1.2\ 10^{1}$
^{238}U	$2.2\ 10^{2}$	$1.5\ 10^{3}$	$1.3\ 10^{2}$	$9.2\ 10^{2}$	$3.6\ 10^{-1}$	$2.5\ 10^{0}$	$2.0\ 10^{2}$	$1.4\ 10^{3}$	$8.2\ 10^{1}$	$5.8\ 10^{2}$	$7.6\ 10^{0}$	$5.3\ 10^{1}$	$6.3\ 10^{1}$	$4.4\ 10^{2}$
^{237}Np	$5.0\ 10^{-2}$	$1.4\ 10^{-1}$	$2.5\ 10^{-2}$	$7.0\ 10^{-2}$	$2.7\ 10^{-2}$	$7.6\ 10^{-2}$	$3.2\ 10^{-2}$	$9.0\ 10^{-2}$	$2.7\ 10^{-2}$	$7.6\ 10^{-2}$	$1.2\ 10^{-2}$	$3.4\ 10^{-2}$	$1.2\ 10^{-2}$	$3.4\ 10^{-2}$
Sub-total	$5.4\ 10^{4}$	$1.6\ 10^{4}$	$3.5\ 10^{4}$	$1.0\ 10^{4}$	$1.8\ 10^{2}$	$9.2\ 10^{2}$	$5.0\ 10^{4}$	$1.5\ 10^{4}$	$2.1\ 10^{4}$	$6.4\ 10^{3}$	$2.2\ 10^{3}$	$7.7\ 10^{2}$	$1.6\ 10^{4}$	$4.8\ 10^{3}$
Total	$7.0\ 10^{4}$		$4.5\ 10^{4}$		$1.1\ 10^{3}$		$6.5\ 10^{4}$		$2.7\ 10^{4}$		$3.0\ 10^{3}$		$2.1\ 10^{4}$	

* Transport model discussed in paragraphs 44-46

** Dosis include contribution of ^{231}Pa

(1) Fresh water collective dose contribution

(2) Ocean collective dose conctribution

repository by a geosphere transport model (paragraphs 44 to 46 and 51 (72)). As indicated (paragraph 51), the results refer to high level waste, enrichment and fabrication wastes and depleted uranium wastes; they are shown in Table 5.a. The results are in good agreement with the sum of the assessments presented in paragraph 74 for high level waste and those for the other three types of wastes given in paragraphs 100, 102 and 106.

77. As expected, for disposal of the unprocessed spent fuel the larger amounts of actinides due to the presence of the entire inventory of uranium and plutonium, results in a somewhat larger radiological impact estimate. However, in view of the great uncertainties involved in the assessments, it is uncertain that the resulting values are really different.

78. It is possible to provide some perspective for the results referring to high level wastes by comparison with the exposures to natural radiation sources. The collective dose commitments from the different fuel cycle strategies, expressed as periods of time during which the exposure of the world population to natural radiation sources would produce the same collective dose, are given in the following table as rounded figures:

Strategy	Equivalent time of exposure to natural radiation sources (minutes/GW(e)y)
1	12 - 37
2	1 - 4
3	2 - 7
4	43 - 62
5	1 - 6
6	5 - 36
7	3 - 17

4.3.3. Estimates of Incomplete Collective Dose Commitments

79. As it was discussed in section 2, the incomplete collective dose commitment is used as a tool in assessments of the future maximum per caput doses from continuing practices, in particular when the exposures are delivered over very long times after the practice has ceased. Incomplete collective dose commitments, being integrated usually over only a few hundred years, provide also information on the fraction of the radiation impact which can be expected in the near future. This information is particularly illustrative if the total radiation impact has been defined over extremely long times.

80. Using 500 years as the integrating period (1, 5), the imcomplete collective dose commitment have been estimated for the time in the future when the integral is maximum, using the approximate technique described in paragraph 11. The results are shown in the following table:

Strategy	Incomplete collective dose commitment (man rem/GW(e)y)
1	$3.7\ 10^{2}$ - $7.9\ 10^{2}$
2	$2.3\ 10^{1}$ - $7.8\ 10^{1}$
3	$3.6\ 10^{1}$ - $1.3\ 10^{2}$
4	$1.1\ 10^{3}$ - $1.6\ 10^{3}$
5	$1.4\ 10^{1}$ - $9.4\ 10^{1}$
6	$1.2\ 10^{2}$ - $5.7\ 10^{2}$
7	$6.2\ 10^{1}$ - $3.5\ 10^{2}$

81. The results shown above are of the same order as the incomplete collective dose commitments reported for important steps of the fuel cycle other than waste management (1, 3). The maximum per caput (average over the population) dose rate expected in the future from the

the waste management of the high level wastes can be assessed, if the practice rate (nuclear power produced) per caput is known. Assuming a value of 1 kW/man, the incomplete collective dose commitments shown in paragraph 80 correspond to maximum per caput dose rates ranging in the order of 0.1 to 1 mrem per year, namely 0.1 to 1 per cent of the dose rate from the natural radiation background. This, however, is a generic assessment subject to variations depending on site-specific conditions of the repositories.

4.3.4. Estimates of Individual Doses in Members of the Public

82. Assessments of individual doses are important because individual dose limitation is one of the basic requirements of radiological protection. It is conceivable that the doses in the most exposed individual (e.g. in the "critical groups") could be higher than the per caput doses (population average doses) presented in the previous paragraph, but the actual values would depend on site-specific conditions. These doses, therefore, should be examined. In this case, obviously, it is not relevant to make the assessments per unit electrical energy, because the main purpose is to compare the estimated doses with appropriate limits.

83. Two assessments, relating to somewhat idealized sites, can be used to indicate the order of magnitude of the individual doses. In one assessment for a salt repository carried out specially for INFCE (72), and which was discussed in relation with transport of nuclides (paragraph 44 to 46), input into circulating waters (paragraph 51) and resulting collective dose commitment (paragraph 76), the individual doses are given directly by the models employed.

84. The repository is assumed to handle the wastes from a fuel cycle required to produce 100 GW(e)y, and it is assumed to be located in a large sedimentary basin, the top of the salt formation being at 250 m below surface. As discussed before, the release assumption involves a geological event creating a fracture, which in connection with existing gradients in the aquifer system results in brine flowing through the repository.

85. The model takes account of waste dissolution, the geosphere transport and the input into biospheric waters, giving the concentrations of the nuclides as a function of time. The dosimetric models can give maximum individual doses and average doses for several population groups.

86. Considering the individual dose in critical groups, only ^{226}Ra and ^{129}I result in non completely negligible doses by the fresh water pathway. The maximum annual effective dose equivalent varies for the different fuel cycles, the biggest being about 3 mrem for fuel cycle strategy N°1, and the lowest 0.02 mrem for fuel cycle strategy N°3. The time of occurrence of all doses was of the order of 10^6 years.

87. Actinide concentrations in river water was found to be negligible for individual dose contributions. Higher concentrations were predicted for ground water, but as the high salt concentration would require at least a 1000 times dilution for drinking, the resulting doses from direct use would be small. The highest annual dose contribution would be given by ^{237}Np (about 0.5 mrem). Actual dilution by fresh water masses would result in three order of magnitude smaller doses.

88. In another study (5), individual doses in the future due to a repository in granite, were assessed for possible sites in Sweden. Nuclide transport from the repository is postulated to occur by ground water flow, and three main pathways were analysed, namely water from a well in the vecinity, arrival into a small lake and arrival into the Baltic Sea. The calculated doses include those from inhalation and from consumption of water and food, as well as those from activity deposited on the ground or in sediments. Maximum individual effective doses were found for the well pathway. The doses are delivered mainly by ^{237}Np and were estimated to be of the order of 10 mrem in a year, after periods exceeding 10^5 years, for a repository of 300 GW(e)y. Considerable conservationism is built in this assessment.

89. It would appear, therefore, that the maximum future doses in critical groups in the vecinity of repositories can be assumed to be very low, of about a few percents of that experienced from the exposure to natural radiation sources.

4.4. Impact of Eventual Disruptive Events on Radiation Safety

90. As it was discussed previously, the basic safety concept of deep geological disposal is the isolation of the high level wastes from man's environments until the potential radiation exposures are reduced by decay to levels compatible with the requirements of radiation protection. Transport of the long-lived nuclides by circulating ground waters was considered in previous sections to occur continuously at very low rates or to be sufficiently probable to be postulated as the "normal" mechanism by which waste nuclides return to man's environment.

91. Other more infrequent events can disrupt the isolation of the wastes. Examples of such events are different forms of volcanogenic transport, seismic displacements, diapirism, human actions, erosion and meteorite impacts. The probability of most of these events is minimized by the adequate site selection for the repository, which takes account of regional tectonics and seismicity, the existence of mineral resources of eventual interest and the adequacy of the host formation.

92. Disruptive events are characterized by their magnitude (activity of nuclides released) and their probability. The contribution of disruptive events to the collective effective dose equivalent caused by the disposal operations depends on the product of magnitude and probability, added for all possible events. In view of the postulated "normal" release of the long-lived nuclides, the occurrences of a disruptive event could be interpreted as a decrease of the "age" of the released mixture and therefore an increase of the normalized collective dose. If the probability of disruptive events is much smaller than the ratio of the potential collective dose of "old" and "young" waste mixtures (Tables 6), the contribution of such events to the collective dose would be negligible in relation to the "normal" contribution discussed previously.

93. A direct expulsion of the waste nuclides to land surfaces or even into the air could result from a meteorite impact or from violent volcanic activity. The probability of a meteorite impact over a 10 km^2 repository, of sufficient energy and momentum, has been estimated to be of the order of 10^{-13} per year (40, 41, 73). Disruption of the repository in a period of 10^3 - 10^4 years would be extremely improbable and the contribution of such events to the collective dose can therefore be

neglected. The probability of direct violent volcanic expulsion in an adequate selected site is of the same order, while combined processes, first transporting the wastes to a shallower depth followed by expulsive events, have even much lower probabilities (40, 41, 73).

94. The probability of human actions which might affect isolation of the wastes has been assessed (41) assuming loss of memory of the repository in about 10^5 years, an increasing future pressure for mining, but a well selected site lacking resources of eventual interest, this being the most important parameter. Even assuming a more rapid (and therefore more realistic) loss of memory, the probability of human actions disrupting isolation in the first 10^4 years is estimated to be of the order of 10^{-6} (41), and, therefore, the contribution of such actions to the collective dose could be neglected.

95. The possibility that a fault could affect the repository is clearly site-specific. However, for tectonically inactive areas, a frequency of 10^{-8} per year has been estimated for evaporitic basins of about 5000 km^2 (41). For an average fault length of about 30 km, the influence area would be about 600 km^2 and the probability of a disruptive event in the first 10^4 years of the repository would be of the order of 10^{-5}. The probability of an actual transport of nuclides from the repository to the surface would be even smaller.

96. Erosion over long periods is a likely process, but the rate is so slow that denuding the waste horizon would require more than 10^6 years, not affecting therefore the estimates of collective dose. The same conclusion applies to other slow geological processes.

97. The conclusions presented above regarding the negligible contribution of disrupting events to the collective dose from waste repositories do not imply that site-specific, or rock-specific, assessments are not required, particularly for satisfying requirements of radiation safety of critical groups. In fact, available data are too few for the sophistication of the fault-tree methods used in safety assessments. Most geological formations are still poorly characterized at the horizons visualized for geological disposal. It has been pointed out (71) that while some estimations could be made for the 10^3 years phase, it is difficult at present to estimate probabilities for rare and long-lasting events in the fault-tree for the 10^6 years phase.

5. RADIOLOGICAL IMPACT FROM MANAGEMENT OF WASTES FROM OTHER FUEL CYCLE STEPS

5.1. Conversion and Enrichment Wastes

98. Uranium ore concentrates produced at the mills are further processed and purified before being converted to uranium dioxide and fabricated into fuel elements. Fuel cycle strategies 1, 2, 6 and 7, require an enrichment step before conversion to uranium dioxide. The major waste from the enrichment operation is UF_6 containing depleted uranium, nominally of about 0.25% ^{235}U. For the purpose of this assessment, it is assumed that this product is converted into a solid form suitable for emplacement in an underground repository, for example as UO_2 packaged in 200 litre drums (9). An additional small waste contribution arises from the conversion of natural uranium into UF_6, estimated to be about 0.4% of the uranium involved (9). Table 6.1 summarizes the nuclide composition of the wastes from conversion and enrichment for the fuel cycle strategies 1, 2, 6 and 7, normalized per GW(e)y of electrical energy produced.

99. The engineered aspects of isolation in this case are of little importance because the drums and the matrix are unlikely to afford any relevant retention time. On the other hand, the geological aspects of isolation (section 4.1.2) would be operative and isolation times of 10^5 to 10^6 years could be expected. This isolation period, however, does not reduce the collective dose contribution, due to the very long half-lives of the nuclides involved.

100. It appears that, in a paradoxal effect, a long isolation increases the resulting collective dose because of the build-up of important daughter radionuclides before passage through circulating waters in the way toward the ocean sediment sink. Using the models discussed in section 4.3, if the materials shown in Table 6.1 were to be released after 10^5 or 10^6 years into surface fresh waters and then go to the ocean, the resulting collective dose commitments would be those shown in Table 6.2. Radium-226, in its passage through fresh water, is the main contributor to the collective dose commitment. These values are equivalent to a few minutes of exposure of the world population to the natural radiation sources. The individual effective dose equivalents are negligible.

5.2. Fuel Fabrication Wastes

101. The fabrication of fuel and of fertile assemblies requires the conversion of fuel materials, whether fresh or recycled, into oxides and the manufacture of sintered pellets. Uranium losses in both operations are assumed to be 0.6%, while for plutonium the losses are 0.3% due to more elaborated handling (9). The thorium component of the cycles 6 and 7 give negligible doses compared to the uranium component.

Table 6.1

Activity composition of conversion and enrichment wastes
(normalized per unit electrical energy (Ci/GW(e)y))

Nuclide	Fuel cycle 1	Fuel cycle 2	Fuel cycle 6	Fuel cycle 7
^{238}U	$5.6\ 10^{1}$	$3.3\ 10^{1}$	$2.3\ 10^{0}$	$1.7\ 10^{1}$
^{235}U	$7.2\ 10^{-1}$	$4.6\ 10^{-1}$	$3.6\ 10^{-2}$	$2.4\ 10^{-1}$
^{234}U a/	$1.5\ 10^{1}$	$1.0\ 10^{1}$	$7.6\ 10^{-1}$	$5.0\ 10^{0}$

a/ Enrichment deplets also ^{234}U. The activity of ^{234}U increases with time, reaching that of ^{238}U. Decay products of ^{234}U build-up, in particular ^{230}Th and ^{226}Ra.

Table 6.2

Conversion and enrichment wastes:
Normalized collective dose commitment
(man rem/GW(e)y)

Nuclide	Fuel cycle 1		Fuel cycle 2		Fuel cycle 6		Fuel cycle 7	
	(1)	(2)	(1)	(2)	(1)	(2)	(1)	(2)
^{226}Ra	$6.6\ 10^{3}$	$2.3\ 10^{4}$	$4.4\ 10^{3}$	$1.4\ 10^{4}$	$2.7\ 10^{2}$	$8.5\ 10^{2}$	$2.2\ 10^{3}$	$7.1\ 10^{3}$
^{230}Th	$3.9\ 10^{2}$	$1.4\ 10^{3}$	$2.6\ 10^{2}$	$7.8\ 10^{2}$	$1.6\ 10^{1}$	$5.0\ 10^{1}$	$1.3\ 10^{2}$	$4.2\ 10^{2}$
^{234}U	$5.3\ 10^{2}$	$1.1\ 10^{3}$	$3.4\ 10^{2}$	$6.2\ 10^{2}$	$2.1\ 10^{1}$	$3.8\ 10^{1}$	$1.7\ 10^{2}$	$3.4\ 10^{2}$
^{235}U	$1.9\ 10^{1}$	$1.9\ 10^{1}$	$1.3\ 10^{1}$	$1.3\ 10^{1}$	$8.7\ 10^{-1}$	$8.7\ 10^{-1}$	$6.3\ 10^{0}$	$6.3\ 10^{0}$
^{238}U	$1.5\ 10^{3}$	$1.5\ 10^{3}$	$8.7\ 10^{2}$	$8.7\ 10^{2}$	$5.5\ 10^{1}$	$5.5\ 10^{1}$	$4.5\ 10^{2}$	$4.5\ 10^{2}$
Total	$9.0\ 10^{3}$	$2.7\ 10^{4}$	$5.9\ 10^{3}$	$1.6\ 10^{4}$	$3.6\ 10^{2}$	$9.9\ 10^{2}$	$3.0\ 10^{3}$	$8.3\ 10^{3}$

(1) Mean isolation time 10^{5} years.

(2) Mean isolation time 10^{6} years.

102. The relevant nuclide composition of the wastes, calculated with the assumptions summarized above, is shown in Table 7, normalized per unit energy generated in each of the fuel cycle strategies. As in the case of conversion and enrichment wastes, the drums and the waste matrix do not ensure a long integrity period but geological isolation should provide periods of the order of 10^5 - 10^6 years before the entry of waste nuclides into surface waters (section 4.1.2). Taking into account the decay over such periods and the collective dose commitment models discussed in section 4.3, the normalized collective dose commitments have been estimated for each fuel cycle:

Strategy	Normalized collective dose commitment (man rem/GW(e)y)
1	$5.1\ 10^1$ - $1.3\ 10^2$
2	$4.0\ 10^1$ - $1.3\ 10^2$
3	$2.4\ 10^1$ - $3.1\ 10^2$
4	$3.3\ 10^1$ - $2.1\ 10^1$
5	$1.7\ 10^1$ - $3.7\ 10^1$
6	$1.2\ 10^1$ - $6.5\ 10^2$
7	$1.7\ 10^0$ - $5.6\ 10^1$

103. The contribution made by the fuel fabrication wastes to the collective dose commitment appears to be small in comparison with the contributions from waste arising from other steps of the fuel cycle. Similarly to what was indicated in section 5.1, it would appear that disposal of fuel fabrication wastes, not containing plutonium, by dumping into the sea would have a smaller radiological impact than their geological disposal. The opposite applies to fuel fabrication wastes containing plutonium, specially in cases where ^{238}Pu is present in large activity.

Table 7

Normalized activity composition of fuel fabrication wastes
(Ci/GW(e)y)

Nuclide	Fuel Cycle						
	(1)	(2)	(3)	(4)	(5)	(6)	(7)
^{233}U	-	-	-	-	-	$1.9\ 10^{1}$	$1.5\ 10^{0}$
^{234}U[a/]	$5.3\ 10^{-1}$	$3.6\ 10^{-1}$	$6.7\ 10^{-3}$	$6.0\ 10^{-2}$	$2.7\ 10^{-2}$	$5.5\ 10^{-2}$	$1.4\ 10^{-2}$
^{235}U	$1.5\ 10^{-2}$	$1.0\ 10^{-2}$	$7.2\ 10^{-5}$	$2.7\ 10^{-3}$	$1.2\ 10^{-3}$	$3.3\ 10^{-4}$	$4.3\ 10^{-3}$
^{238}U	$7.4\ 10^{-2}$	$5.3\ 10^{-2}$	$6.7\ 10^{-3}$	$6.0\ 10^{-2}$	$2.7\ 10^{-2}$	$6.1\ 10^{-4}$	$1.2\ 10^{-4}$
^{238}Pu	-	$2.4\ 10^{2}$	$1.7\ 10^{3}$	-	$4.4\ 10^{1}$	-	-
^{239}Pu	-	$1.9\ 10^{1}$	$1.2\ 10^{2}$	-	$1.4\ 10^{1}$	-	-
^{240}Pu	-	$2.2\ 10^{1}$	$1.7\ 10^{2}$	-	$5.7\ 10^{1}$	-	-
^{241}Pu	-	$4.9\ 10^{3}$	$1.4\ 10^{4}$	-	$3.8\ 10^{3}$	-	-
^{242}Pu	-	$1.1\ 10^{-1}$	$4.9\ 10^{-1}$	-	$7.1\ 10^{-1}$	-	-

a/ For fuel cycles 1 and 2 during enrichment ^{234}U is also enriched to a somewhat larger degree than ^{235}U.

104. Both the incomplete collective dose commitment and the maximum per caput doses in the future, from fuel fabrication wastes, are negligible compared with the corresponding values from wastes arising in other steps of the fuel cycle.

5.3. Discarded Depleted Uranium from Reprocessing

105. The reference fuel cycles considered in this document (9) assume that depleted uranium, which is a reprocessing waste in some of the fuel cycles, is converted into UO_2, packed into 200 l drums and disposed into geological repositories. The relevant nuclide composition is shown in Table 8.1.

106. The same considerations made for depleted uranium as enrichment wastes (paragraphs 99 and 100) apply here. Using the models discussed in section 4.2, and assuming that the materials are released into fresh waters after periods of 10^5-10^6 years, proceeding then to the ocean, the resulting collective dose commitments would be those shown in Table 8.2.

6. MISCELLANEOUS CONTRIBUTIONS TO THE COLLECTIVE DOSE COMMITMENT

107. The management of wastes arising from the nuclear fuel cycle involves some occupational exposures and the release of effluents to the environment. Both these contributions are expected to be small compared with the occupational exposures and the effluent releases from other steps of the fuel cycle. As some concepts of waste management have not been yet practiced at the required scale, there is actually no experience on which to base the assessments of these contributions from the management of waste.

Table 8.1

Activity composition of depleted uranium from reprocessing
(normalized per unit electrical energy (Ci/GW(e)y))

Nuclide	Fuel cycle 2	Fuel cycle 3	Fuel cycle 5
^{234}U	$8.6\ 10^{0}$	$1.9\ 10^{-1}$	$2.0\ 10^{1}$
^{235}U	$1.1\ 10^{-1}$	$6.0\ 10^{-4}$	$2.2\ 10^{-1}$
^{238}U	$3.7\ 10^{0}$	$6.7\ 10^{-2}$	$2.4\ 10^{1}$

Table 8.2

Depleted uranium from reprocessing:
Normalized collective dose commitment
(man rem/GW(e)y)

Nuclide	Fuel cycle 2		Fuel cycle 3		Fuel cycle 5	
	(1)	(2)	(1)	(2)	(1)	(2)
^{226}Ra	$2.0\ 10^{3}$	$1.9\ 10^{3}$	$4.4\ 10^{1}$	$3.5\ 10^{1}$	$5.7\ 10^{3}$	$1.1\ 10^{4}$
^{230}Th	$1.2\ 10^{2}$	$1.1\ 10^{2}$	$2.6\ 10^{0}$	$2.1\ 10^{0}$	$3.4\ 10^{2}$	$6.4\ 10^{2}$
^{234}U	$1.1\ 10^{1}$	$7.3\ 10^{0}$	$3.4\ 10^{0}$	$1.6\ 10^{0}$	$4.5\ 10^{2}$	$4.9\ 10^{2}$
^{235}U	$2.9\ 10^{0}$	$2.9\ 10^{0}$	$1.6\ 10^{-2}$	$1.6\ 10^{-2}$	$5.8\ 10^{0}$	$5.8\ 10^{0}$
^{238}U	$9.7\ 10^{1}$	$9.7\ 10^{1}$	$1.8\ 10^{0}$	$1.8\ 10^{0}$	$6.3\ 10^{2}$	$6.3\ 10^{2}$
Total	$2.2\ 10^{3}$	$2.1\ 10^{3}$	$5.1\ 10^{1}$	$4.0\ 10^{1}$	$7.1\ 10^{3}$	$1.3\ 10^{4}$

(1) Mean isolation time 10^{5} years.

(2) Mean isolation time 10^{6} years.

108. There is some information on the collective dose from waste processing in power reactors (1). This contribution appears to be about 7% for the total occupational collective dose found for the operation of the reactors, which is about 1 man rem/MW(e)y for the power reactors presently in operation (1). The occupational collective dose from waste processing in power reactors is, therefore, estimated to be about 70 man rem/GW(e)y for all the fuel cycles. This value may decrease in the future, as the level of ambition for radiation protection appears to be increasing for many national authorities.

109. Regarding repositories for high level wastes, there is no actual experience on occupational doses. Some projections are however possible, under asusmptions concerning the number of workers in the repository and ancillary facilities. Assuming this number to be of the order of 10^3 (75) it is obvious that the collective dose will be smaller than 5000 man rem in a year because of theindividual dose limits. Furthermore, in many occupations involving radiation exposures, the mean dose is in the order of 10% of the dose limits (1). If the same distribution would apply to occupational exposures in repository operations, the collective dose would be of the order of 500 man rem in a year. Assuming 30 years of operation and a total disposal of the order of 50 GW(e)y (75), the occupational collective dose would certainly be smaller than 3000 man rem/GW(e)y, possibly by a factor of 10, not being, therefore, a major contribution to the total collective dose from waste management and disposal.

110. The separation of ^{85}Kr in the reprocessing plants and its surface storage would not contribute to the collective dose, unless an accidental release occurs.

As the release without retention of ^{85}Kr implies a collective dose commitment of 400 man rem/GW(e)y (effective dose, calculated from published tissue dose commitments (1)), an accidental release could not exceed that collective dose contribution. The possibility of significant individual doses requires careful safety considerations in the design of the storage.

111. The effluents resulting from waste management operations (5) are expected to result in much smaller collective dose commitments than those from the effluents from the long term disposal of the wastes.

7. SUMMARY

112. The radiological impacts from the management of wastes from the nuclear fuel cycle have been estimated for several alternative fuel cycle strategies. The impacts are expressed as collective effective dose equivalent commitments, and therefore all qualifications attached to this quantity (section 2) should be kept in mind when evaluating the results, which are summarized in Table 9.

113. The uncertainties qualifying the values shown in Table 9 imply that they even could not differ between themselves. Mill tailings make an important contribution, which depends on the uranium requirements for each reference fuel cycle, being the largest for once-through cycles. Disposal of high level wastes or spent fuel is also an important contribution, usually larger for once-through cycles where the entire actinide inventory is disposed off. Although at present conversion and enrichment tails are not considered wastes, they have been assumed to be wastes in the reference

cycles. In this case, their relative contribution is significant, for fuel cycles using enriched uranium.

114. The values shown in Table 9 should be seen in perspective. The totals for waste management and disposal are of the same order of magnitude as the collective dose commitments from occupational and public exposures arising from the operation of the nuclear fuel cycles. On the other hand, if expressed in equivalent periods of time during which the world population would receive the same collective doses from the natural radiation sources, the totals range between about 5 and 110 minutes of natural irradiation per GW(e)y, as shown in Table 10.

115. The incomplete collective dose commitments from waste management and disposal, assessed by integrating the collective dose rate over a fixed period of time (usually selected as 500 years) at a time when the integral is maximum (section 2), are also comparable with the corresponding quantity arising from the operation of the fuel cycles (sections 3, 2) (1). The maximum per caput doses in the future are small, usually a small fraction of the relevant dose limits.

116. The maximum future doses in the critical groups in the vecinity of the repositories will be very low, of about a few percents of that experienced from the exposure to natural radiation sources.

Table 9

Normalized collective dose commitments from waste management and disposal

(man rem/GW(e)y)

	Strategy													
	(1)		(2)		(3)		(4)		(5)		(6)		(7)	
Mill tailings	$3.7\ 10^{4}$		$2.2\ 10^{4}$		$2.2\ 10^{2}$		$3.3\ 10^{4}$		$1.3\ 10^{4}$		$1.3\ 10^{3}$		$1.0\ 10^{4}$	
Waste from fuel cycle steps following reactor operation	$8.9\ 10^{3}$	$2.7\ 10^{4}$	$9.7\ 10^{2}$	$2.8\ 10^{3}$	$1.8\ 10^{3}$	$5.2\ 10^{3}$	$3.1\ 10^{4}$	$4.5\ 10^{4}$	$6.8\ 10^{2}$	$4.6\ 10^{3}$	$3.6\ 10^{3}$	$2.6\ 10^{4}$	$2.5\ 10^{3}$	$1.2\ 10^{4}$
Depleted uranium from reprocessing	-	-	$2.1\ 10^{3}$	$2.2\ 10^{3}$	$4.0\ 10^{1}$	$5.1\ 10^{1}$	-	-	$1.3\ 10^{4}$	$7.1\ 10^{3}$		-	-	-
Waste from conversion and enrichment	$2.7\ 10^{4}$	$9.0\ 10^{3}$	$1.6\ 10^{4}$	$5.9\ 10^{3}$	-	-	-	-	-	-	$9.9\ 10^{2}$	$3.6\ 10^{2}$	$8.3\ 10^{3}$	$3.0\ 10^{3}$
Waste from fuel fabrication	$5.1\ 10^{1}$	$1.3\ 10^{2}$	$4.0\ 10^{1}$	$1.3\ 10^{2}$	$2.4\ 10^{1}$	$3.1\ 10^{2}$	$3.3\ 10^{1}$	$2.1\ 10^{1}$	$1.7\ 10^{1}$	$3.7\ 10^{1}$	$1.2\ 10^{1}$	$6.5\ 10^{2}$	$1.7\ 10^{0}$	$5.6\ 10^{1}$
Miscellaneous contributions	$3.7\ 10^{2}$		$3.7\ 10^{2}$		$3.7\ 10^{2}$		$3.7\ 10^{2}$		$3.7\ 10^{2}$		$3.7\ 10^{2}$		$3.7\ 10^{2}$	
Total for waste management and disposal	$7.3\ 10^{4}$	$7.3\ 10^{4}$	$4.1\ 10^{4}$	$3.3\ 10^{4}$	$2.5\ 10^{3}$	$6.2\ 10^{3}$	$6.4\ 10^{4}$	$7.8\ 10^{4}$	$2.7\ 10^{4}$	$2.5\ 10^{4}$	$6.3\ 10^{3}$	$2.9\ 10^{4}$	$2.1\ 10^{4}$	$2.5\ 10^{4}$

Table 10

Time of exposure of the world population to the natural radiation sources producing collective dose equal to the contributions of waste management and disposal

(minutes per GW(e)y)

	Strategy													
	(1)		(2)		(3)		(4)		(5)		(6)		(7)	
Mill tailings	51		30		0.3		46		18		2		14	
Waste from fuel cycle steps following reactor operation	12	37	1	4	2	7	43	62	1	6	5	36	3	17
Depleted uranium from reprocessing	-	-	3	3	<1	<1	-	-	18	10	-	-	-	-
Waste from conversion and enrichment	37	12	22	8	-	-	-	-	-	-	1	<1	11	4
Waste from fuel fabrication	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	1	1	1	1
Miscellaneous contributions	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	1	1	1	1
Total for waste management and disposal	102	102	58	47	5	10	91	110	39	36	10	41	30	37

ATTACHMENT 1

FISSION PRODUCT ARRIVAL CURVES

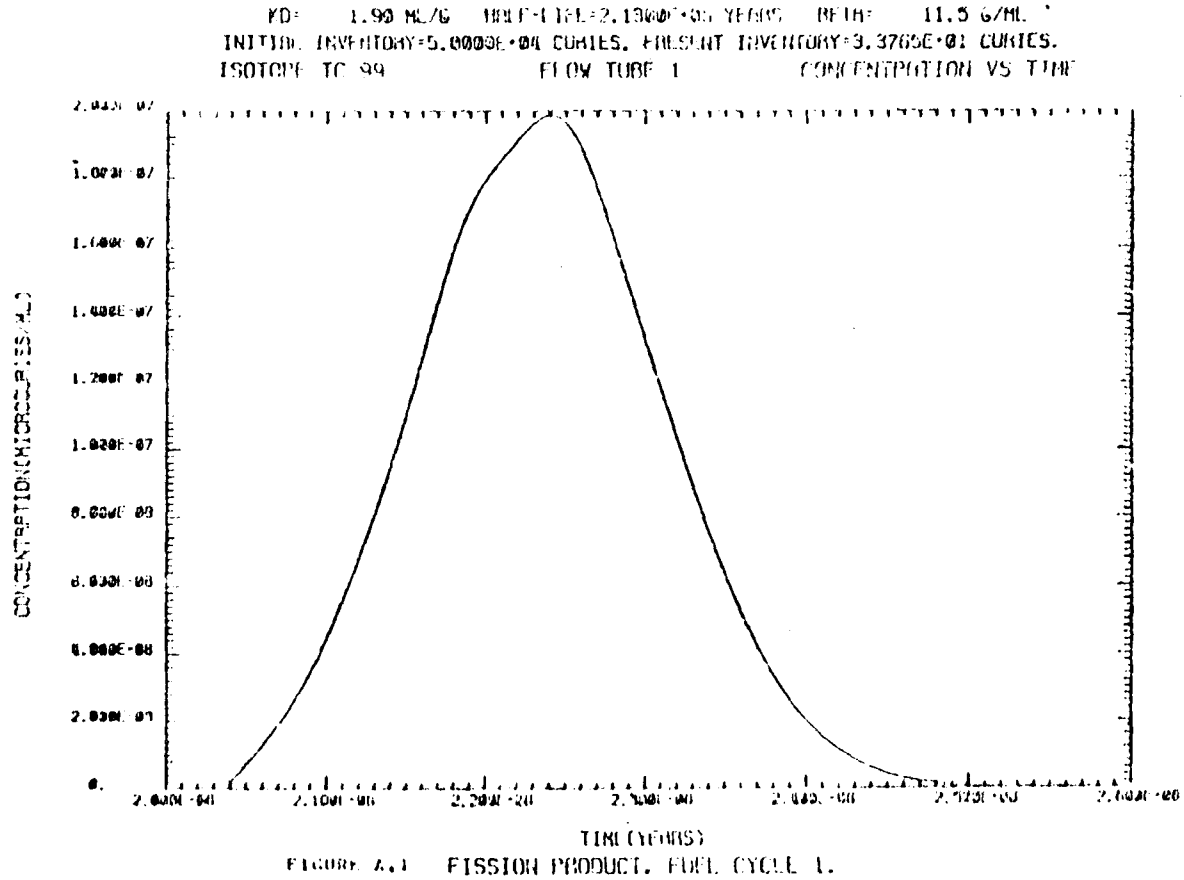


FIGURE A.1 FISSION PRODUCT, FUEL CYCLE 1.

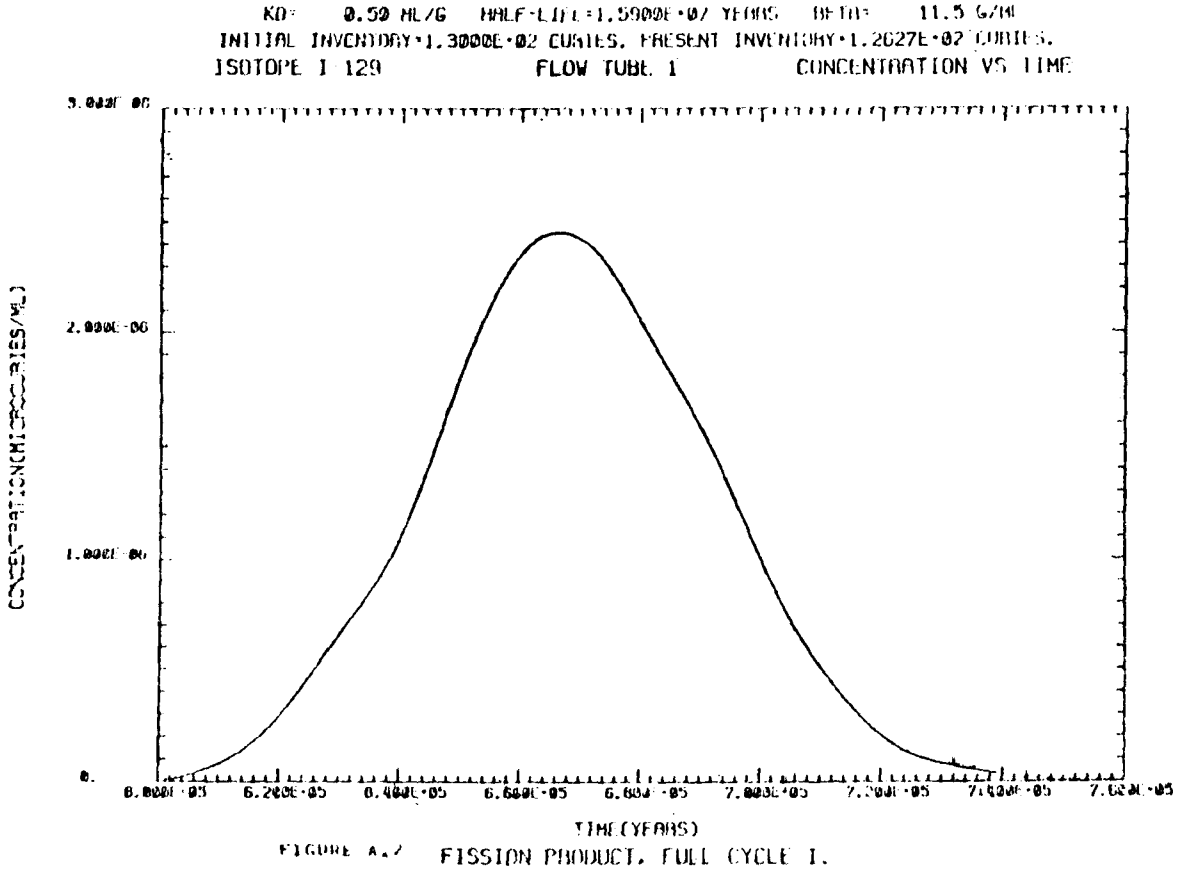
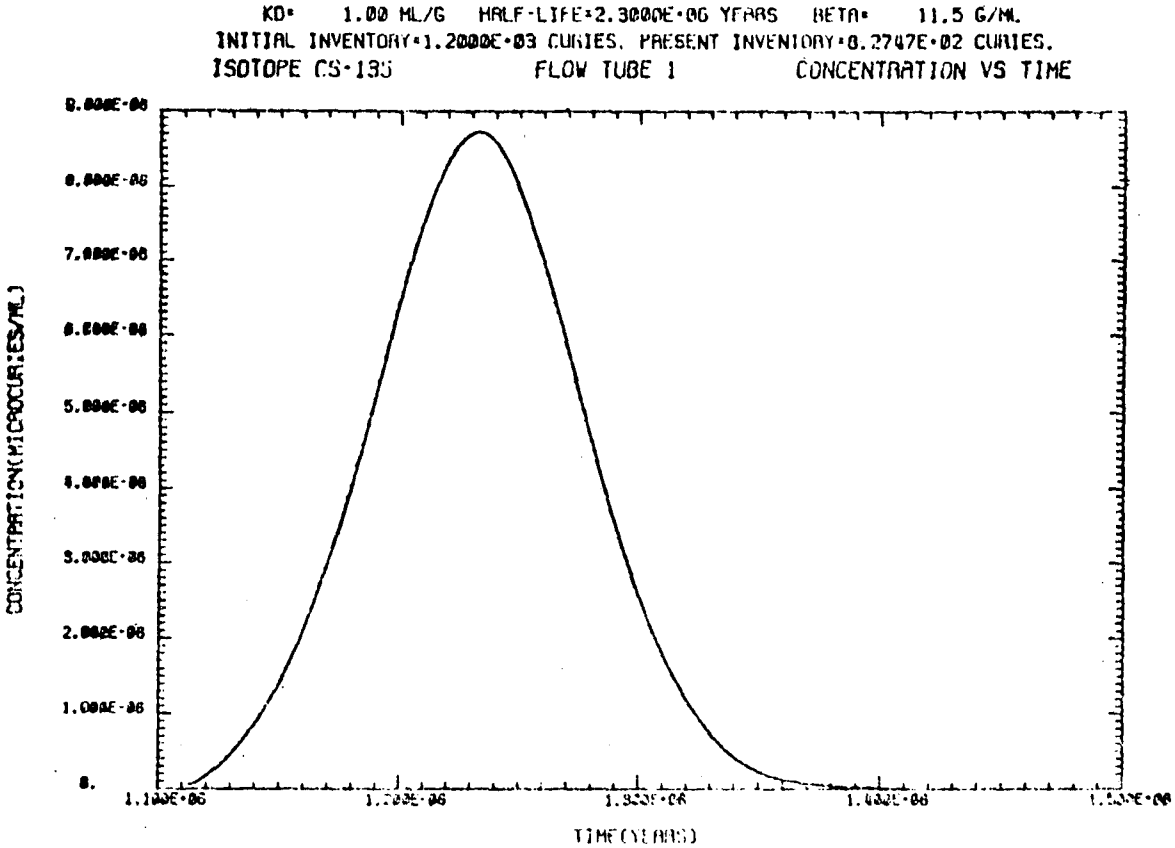


FIGURE A.2 FISSION PRODUCT, FULL CYCLE 1.



FISSION PRODUCT, FUEL CYCLE 2.

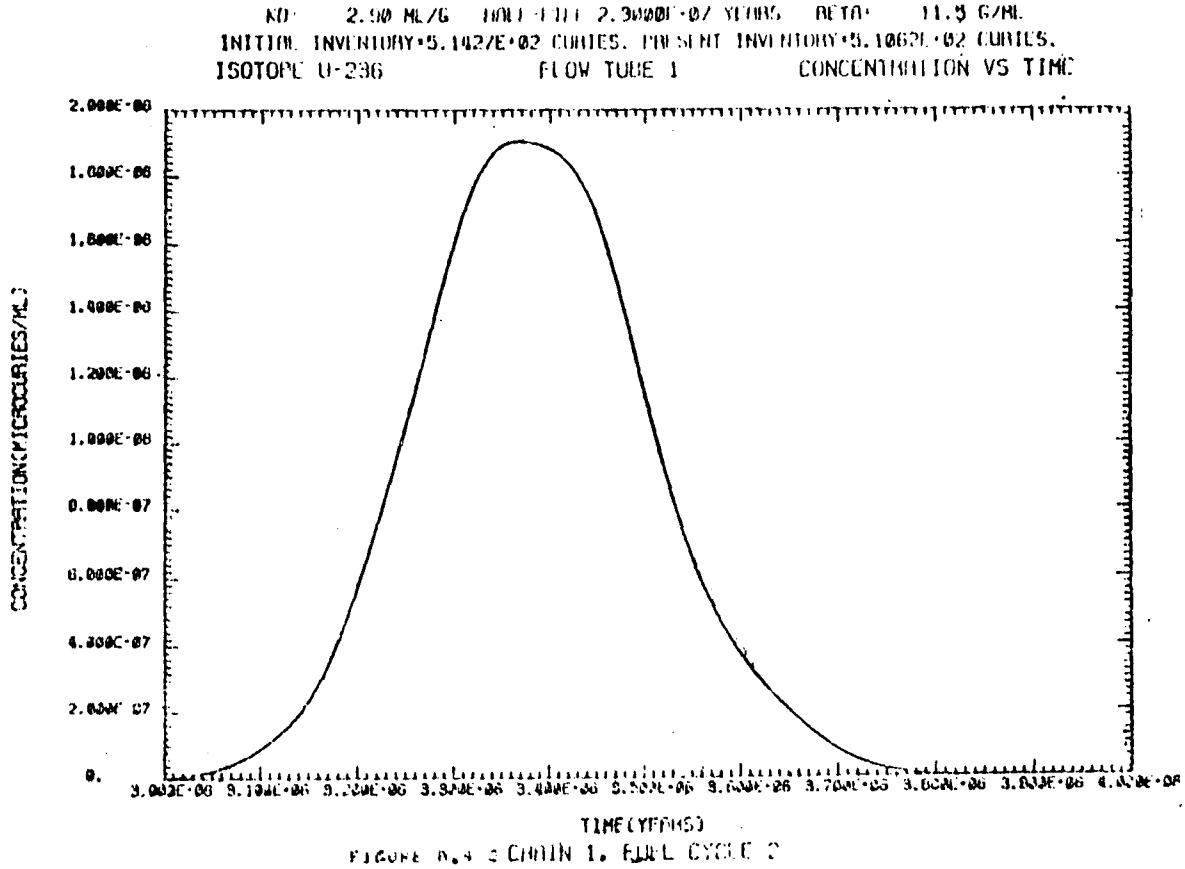
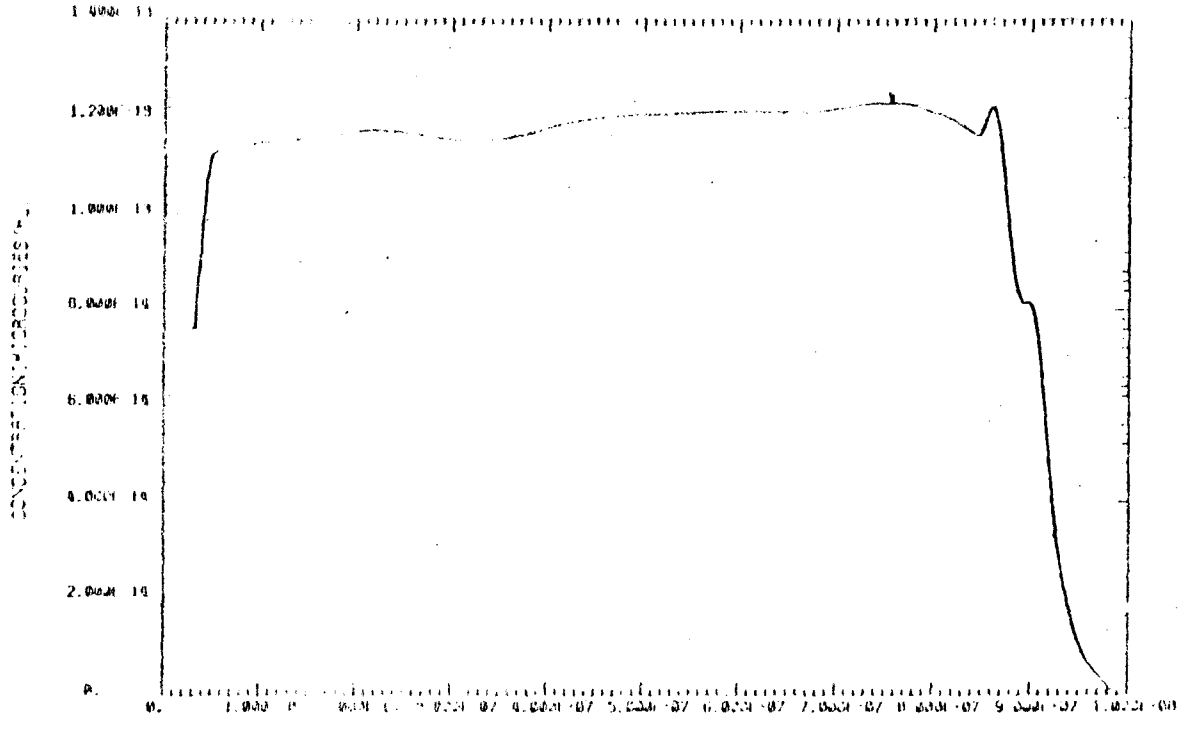
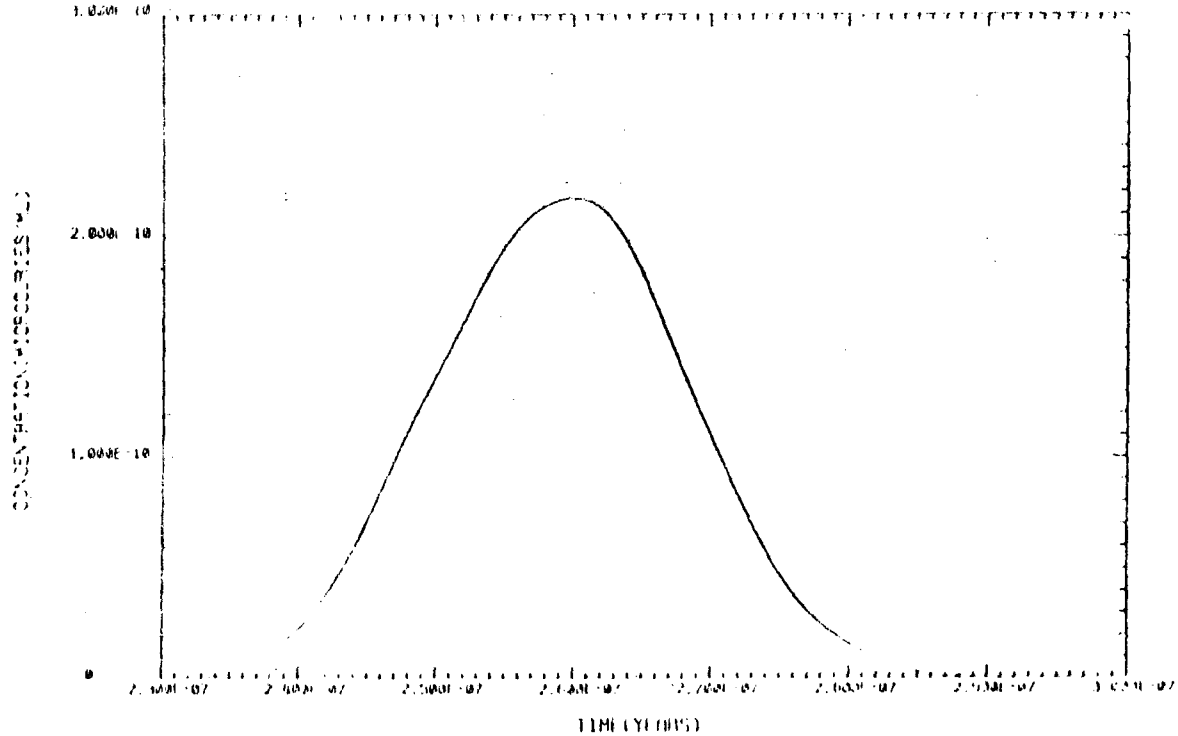


FIGURE B.4 : CHAIN 1, FUEL CYCLE 2



TIME (YEARS)

FIGURE C.5 CHAIN 1, FUEL CYCLE 1



TIME (YEARS)

FIGURE C.6 CHAIN 2, FUEL CYCLE 1

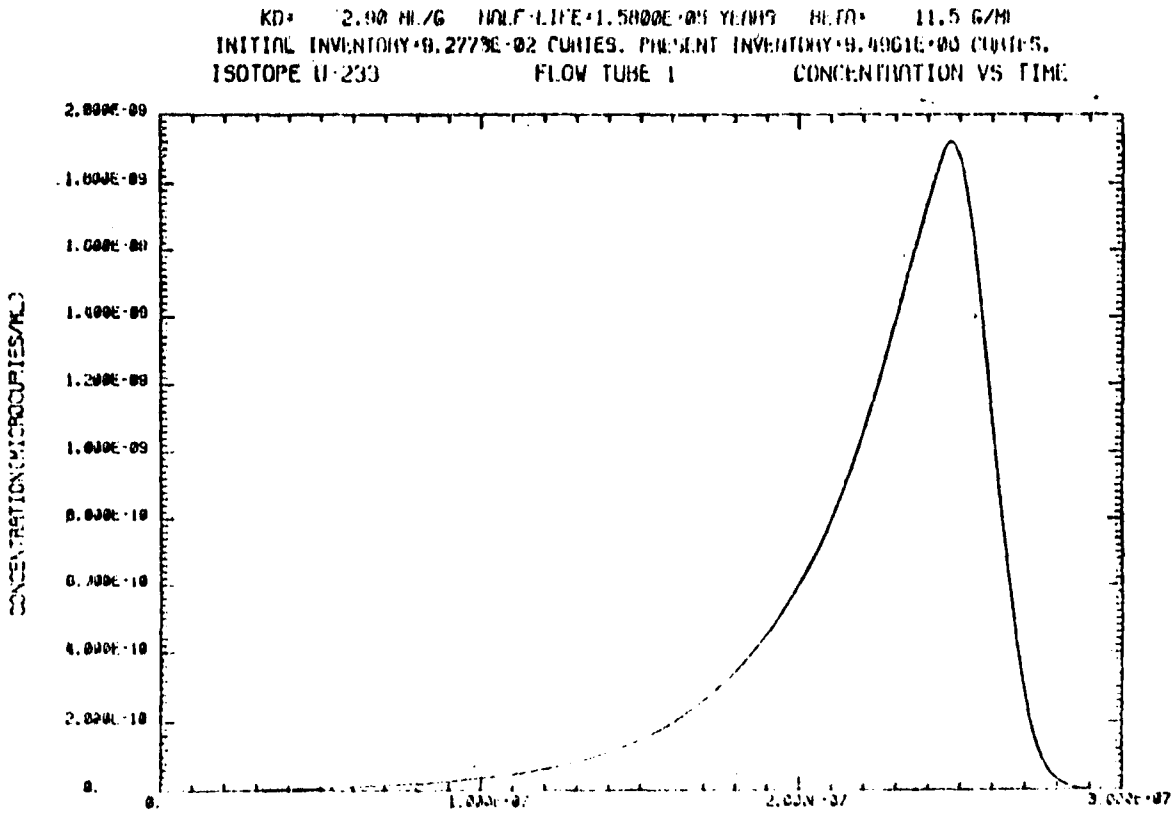


FIGURE B.7 CHAIN 2, FUEL CYCLE 4

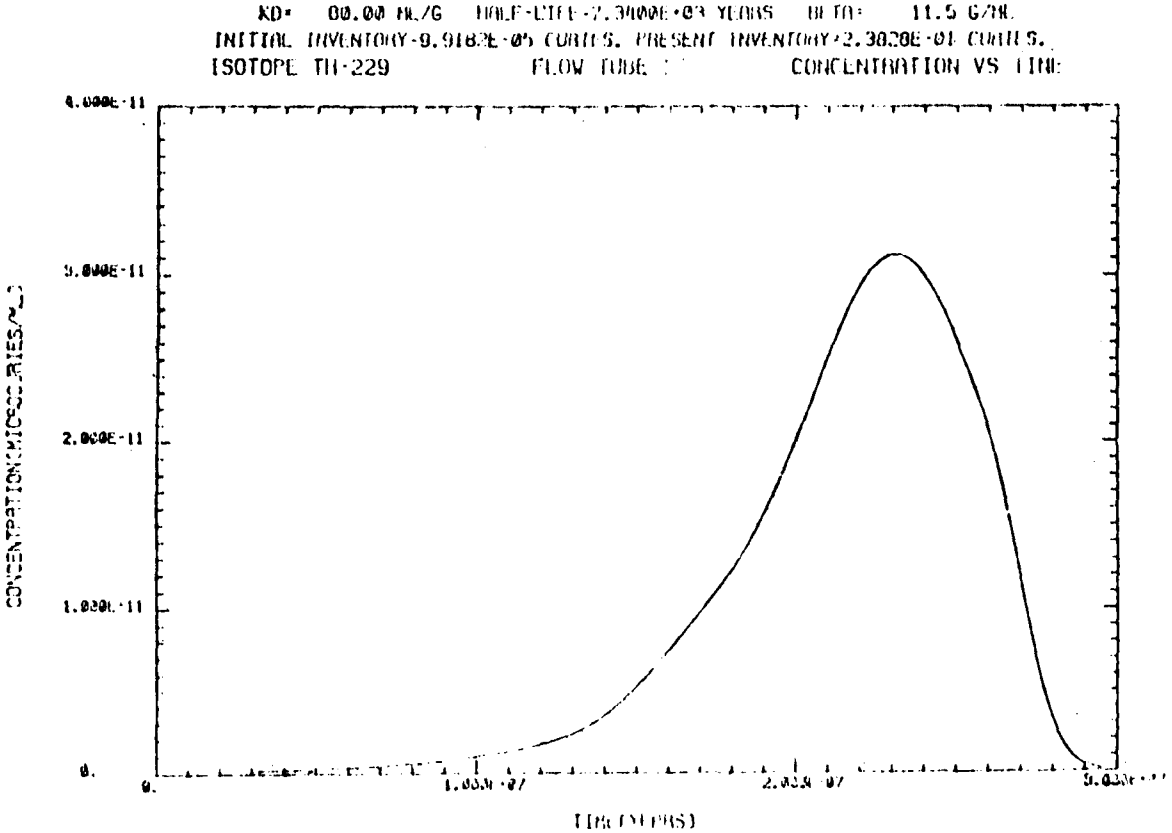


FIGURE B.8 CHAIN 2, FUEL CYCLE 4

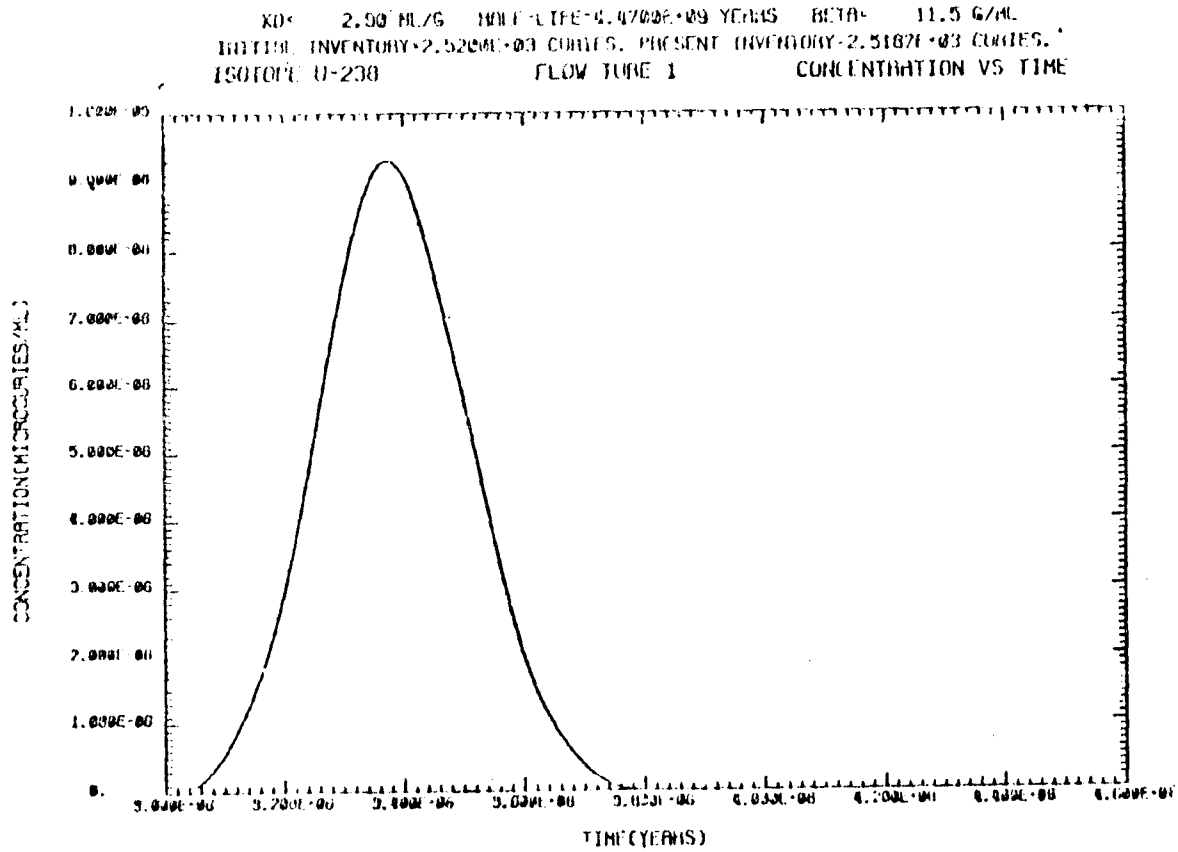


FIGURE E.9 CHAIN 3, FUEL CYCLE 5

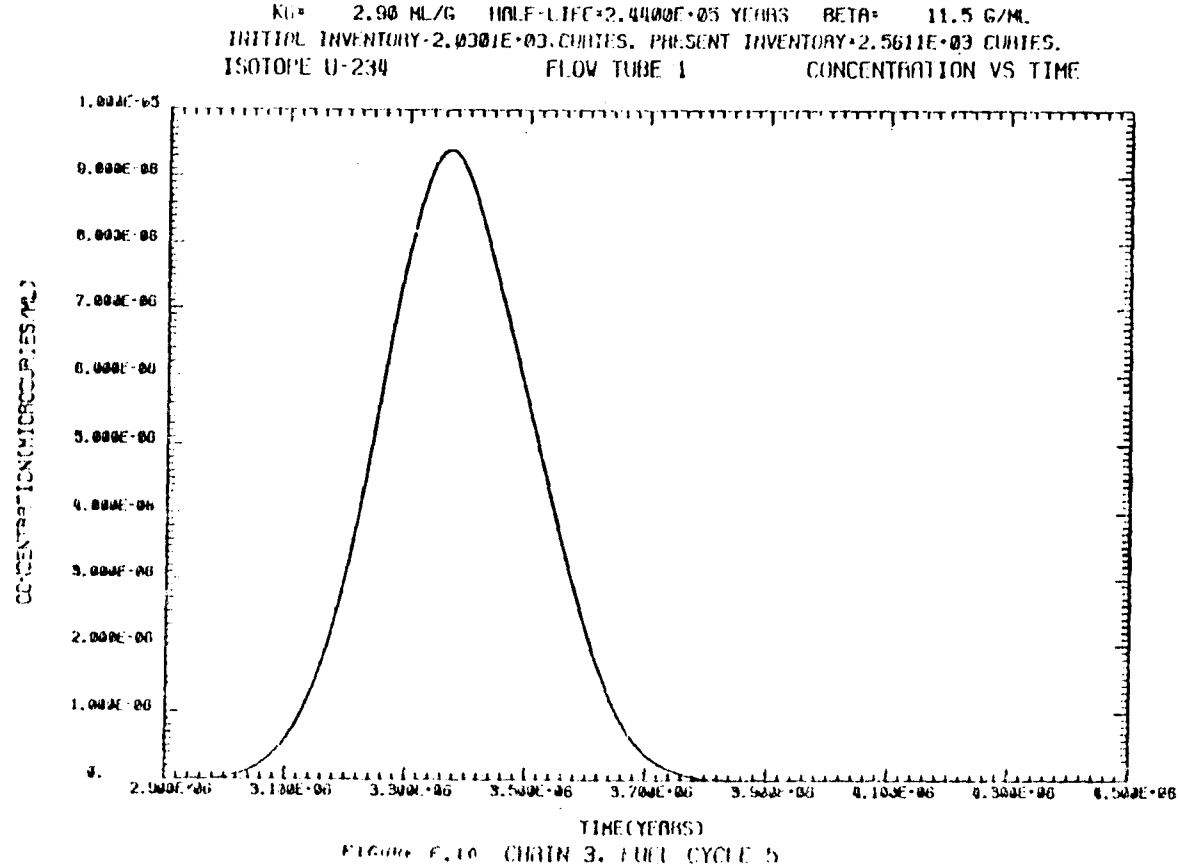
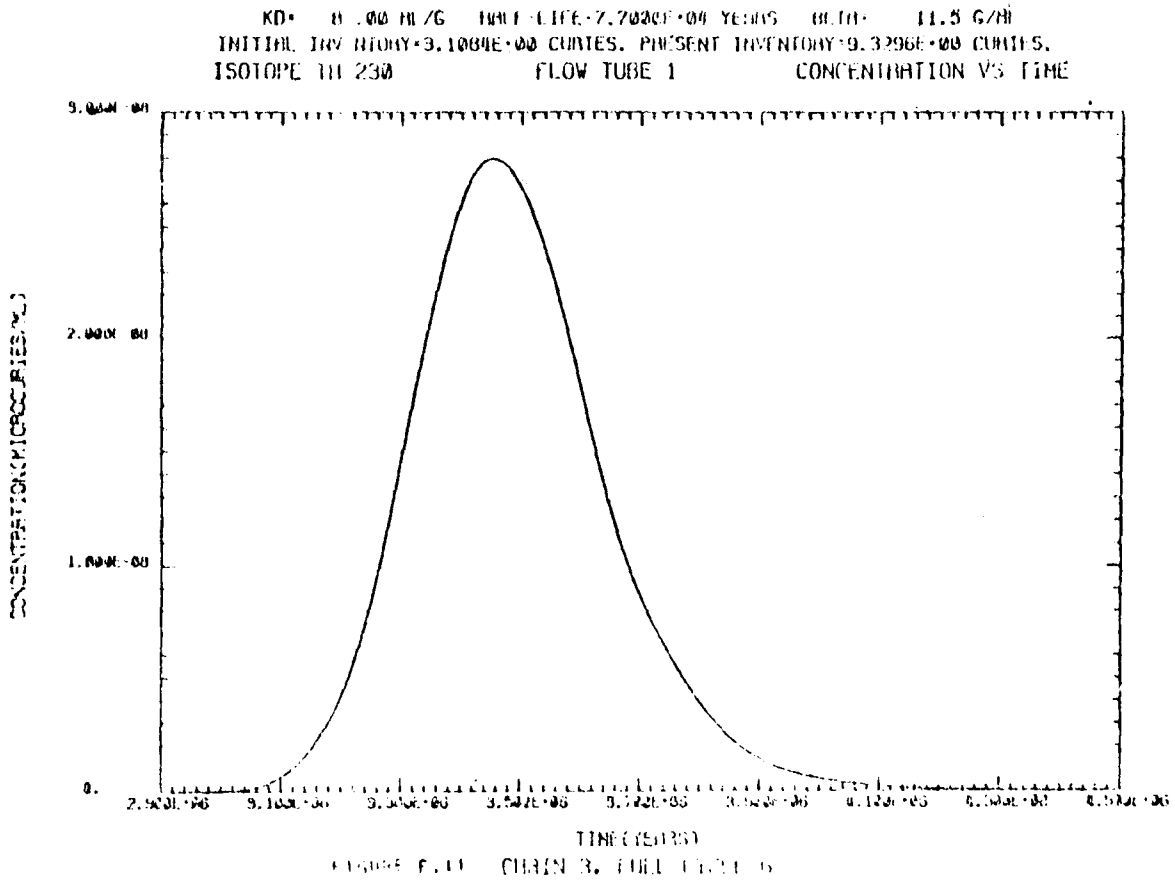
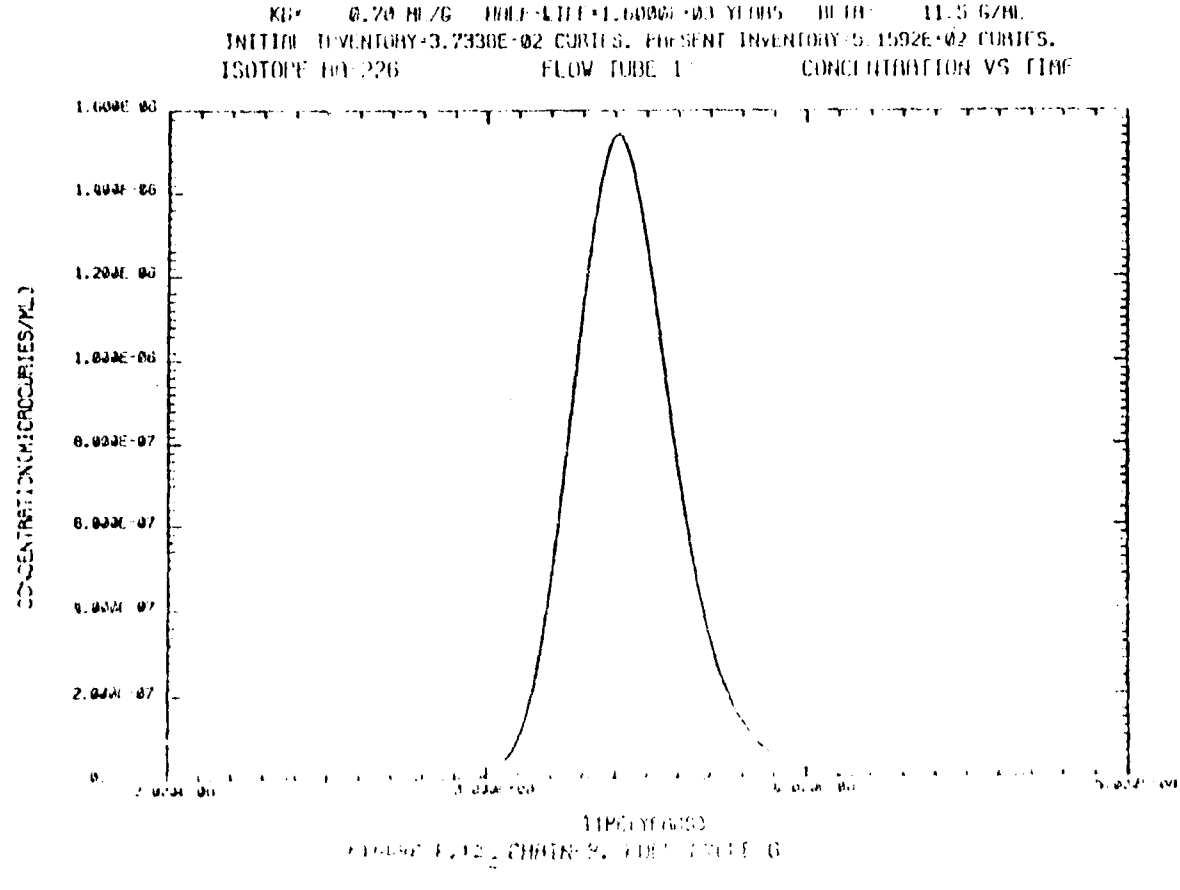


FIGURE E.10 CHAIN 3, FUEL CYCLE 5


KD= 0.00 ML/G HALF-LIFE=7.7000E+04 YEARS DENS= 11.5 G/ML
INITIAL INVENTORY=3.1000E+00 CURIES. PRESENT INVENTORY=9.3296E+00 CURIES.
ISOTOPE TH 230 FLOW TUBE 1 CONCENTRATION VS TIME
CONCENTRATION(MICROCURIES/ML)
TIME(YEARS)


KD= 0.70 ML/G HALF-LIFE=1.6000E+03 YEARS DENS= 11.5 G/ML
INITIAL INVENTORY=3.7330E-02 CURIES. PRESENT INVENTORY=3.1592E-02 CURIES.
ISOTOPE TH 226 FLOW TUBE 1 CONCENTRATION VS TIME
CONCENTRATION(MICROCURIES/ML)
TIME(YEARS)

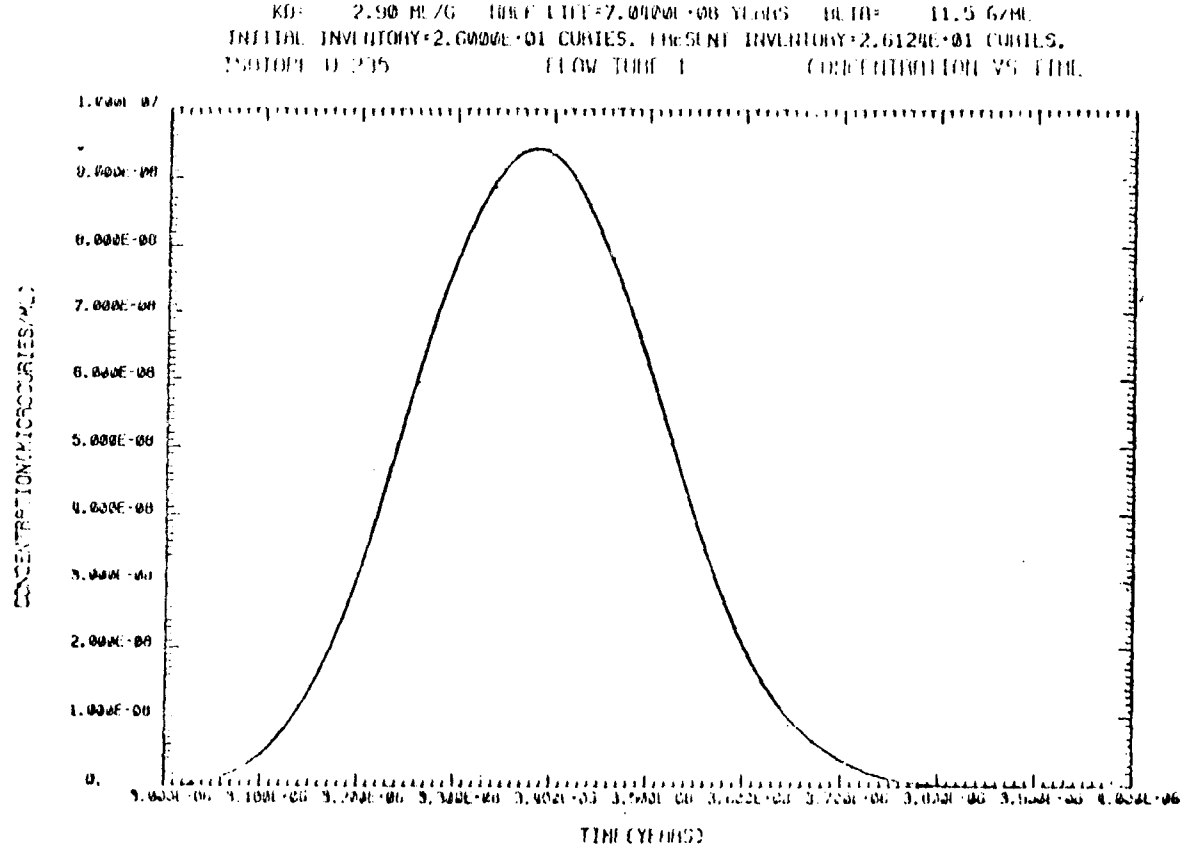


FIGURE 6.13 CHAIN 4, FUEL CYCLE 2

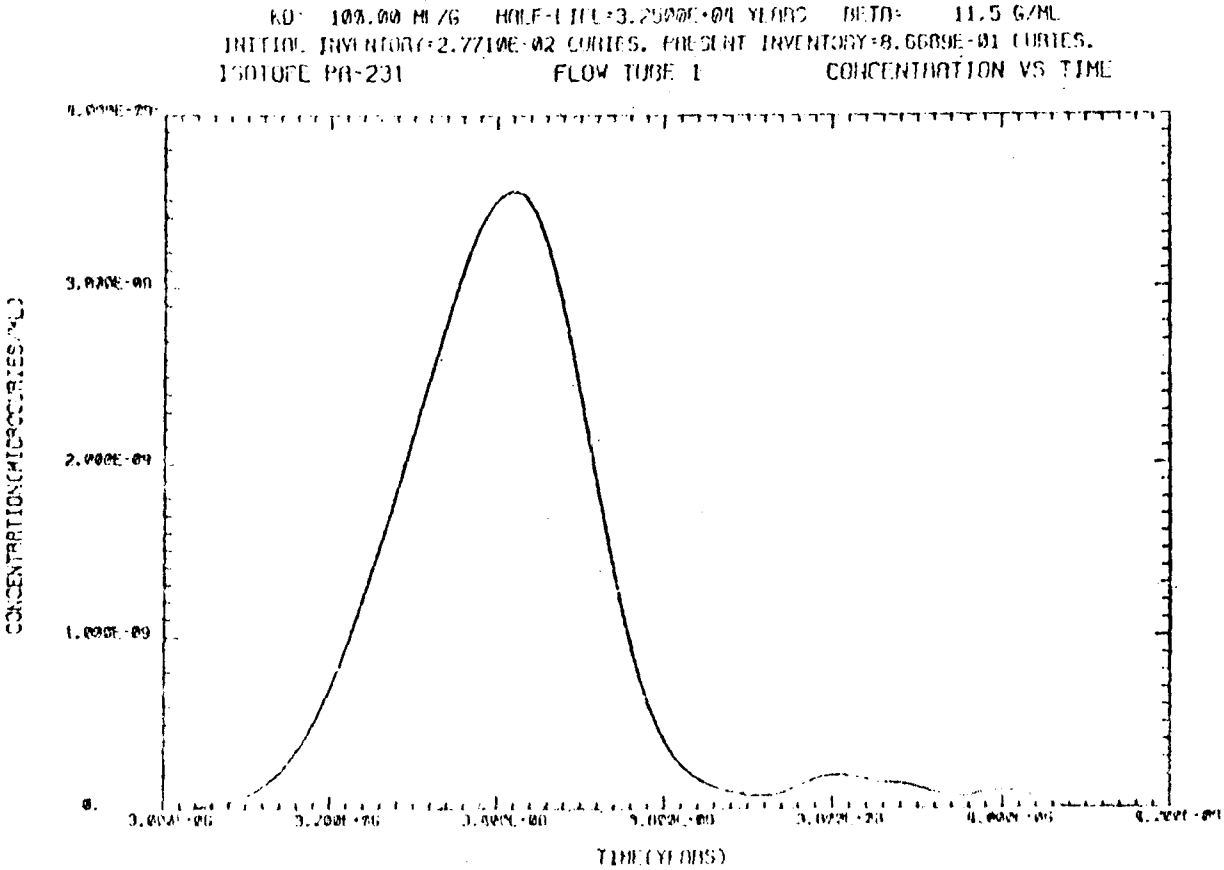


FIGURE 6.14 CHAIN 4, FUEL CYCLE 2

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