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SOURCES AND EFFECTS OF IONIZING RADIATION

United Nations Scientific Committee
on the Effects of Atomic Radiation

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ANNEX D

Radioactive contamination due to nuclear power production

CONTENTS

Introduction

1. The use of nuclear reactors to generate electricity has continued to increase since the Committee's first attempt to assess the radioactive releases from the nuclear fuel cycle and the resulting dose commitment (140). The total installed nuclear electric generating capacity in the world in 1976 was 79.8 GW from 188 reactors operating in 19 countries (66). The predicted capacity for 2000 is 2000 GW, about half the value predicted in the 1972 report of the Committee.

2. The nuclear fuel cycle consists of the process of mining and milling of uranium, conversion to fuel material, usually including enrichment in the isotope ^{235}U, fabrication of fuel elements, utilization of the fuel in nuclear reactors, reprocessing of spent fuel and recycled utilization of recovered fissile materials, transportation of material between fuel-cycle installations, and disposal of radioactive wastes.

3. Almost all the radioactivity associated with the fuel cycle is present in stored, spent fuel elements and in well contained fractions separated from the fuel during the reprocessing operations. However, at each step of the fuel cycle, releases of small quantities of radioactive material into the environment may occur. Most of the radionuclides released into the environment are only of local or regional concern, because their half-lives are short compared to the time required for dispersion to greater distances. Some radionuclides, on the other hand, having longer half-lives or being more rapidly dispersed, can become globally distributed.

4. The Committee is interested in assessing the dose commitments due to releases of radioactive material from all fuel-cycle operations. As the scale of these operations is related to the nuclear installed capacity they serve, it seems reasonable to express such assessments in terms of the dose commitment (or collective dose commitment) per unit of electrical energy generated (MW(e) y).

5. The total collective dose commitment to the world population may be assessed by estimating the contribution from four components: the occupationally exposed group, the local population, the regional or intermediate population, and the world population. This Annex deals only with the collective doses incurred through exposure of the public, while the occupational contributions are analysed in Annex E. The local and regional contributions to the collective dose commitment are discussed separately for each component of the fuel cycle; the global contributions made by nuclides of world-wide distribution are discussed for the cycle as a whole. The handling of solid wastes generated in the nuclear fuel cycle is considered in chapter V of this Annex.

6. Doses and collective dose commitments to local and regional populations can be estimated satisfactorily given adequate knowledge of the effluents, the transfer factors in the routes by which man is exposed and the results of properly designed environmental monitoring programmes. However, the data based on measurements available to the Committee are insufficient for the purpose of assessing all collective dose commitments, especially global collective dose commitments, from all fuel cycle operations, and some generalizations concerning effluent release rates and environmental transfer factors have had to be made. Monitoring programmes are usually confined to local areas and are frequently planned to verify compliance with applicable regulations. More detailed measurements are seldom carried out and in many situations are not even feasible. In some cases, however, monitoring can contribute to improvements in the assessment of model parameters, monitoring and modelling being in these cases complementary procedures.

7. The Committee has reviewed information relating to both approaches. Since the 1972 report, a number of national and international symposia have dealt with various aspects of environmental dispersion mechanisms and of population exposures (46, 63, 67, 69, 149, 150). A wide range of other publications have contributed to increased knowledge of environmental transport and improved estimates of radiation doses. In addition, environmental and safety studies for specific fuel-cycle installations and a few comprehensive reviews of the entire fuel cycle have been prepared (148, 151, 153, 156, 157, 158, 162).

8. The discussion of the effluents of nuclear power installations and the collective dose estimates are, as far as possible, based on operating experience. Included are data on releases in all periods of operation, including those associated with abnormal operation or with periods in which there is no electricity generated or fuel reprocessed. The averages, therefore, do not represent typical release rates during normal operation, but instead reflect the overall nuclear power experience, normalized to the total amount of generated electricity. Where operating experience is insufficiently documented, some generalizations regarding effluent release rates and the environmental transfer parameters are made in order to make a complete assessment of the collective doses from nuclear fuel-cycle operations. It must be stressed that such generalizations will always lead to less accurate estimates of the dose commitments than assessments based directly on studies of particular releases and the particular environment into which the releases are made. Moreover, in the future, improved practices may give results that are considerably different from those derived from current and past experience. Therefore, any application of the assessments given here to more specific situations or any extrapolation to the future must recognize these inherent limitations.

9. In addition to the small releases of radioactive material during normal operation the possibility exists that additional amounts of radioactive materials may be accidentally discharged. A number of studies have considered the potential composition and amounts of accidental releases (61). The assessment of population exposure due to long-term nuclear power programmes requires the estimation of the expected contribution of such events. Recent evaluations of their probability make it possible to assess the expectation of collective dose from the release of radioactive material due to accidents in reactors, as discussed in the appendix.

I. MINING AND MILLING

10. Uranium is obtained from ore mined in several countries throughout the world. The countries with the largest production during 1975 were Canada, France, South Africa and the United States of America. Reserves of varying quantity and grades, and mining operations of varying scale are reported in several other countries (110).

11. Uranium ore has been obtained mainly from underground mines, although large amounts are now coming from open-pit mines. In the United States, 70 per cent of the ore production in 1973 came from 33 open-pit mines and the remainder from 122 underground mines (14). In South Africa, uranium is extracted from gold-mine tailings (117).

12. Since the ore normally contains less than 0.2 per cent uranium, processing mills are usually located in the mining areas to avoid transportation of large quantities of what is ultimately waste material. In some cases, however, ore is treated locally by heap-leaching, and precipitated pre-concentrates are then transported to the mills. In the mills, the ore is blended and crushed, then wet ground and transferred as a slurry to leaching tanks. According to the ore characteristics, either a sulphuric acid or a sodium carbonate alkaline leach process is utilized. The soluble uranium is recovered by solvent extraction, or, in the case of the alkaline leach process, by precipitation. The uranium concentrate is calcined and pulverized, forming the final product of the mill, the semi-refined compound called yellowcake.

A. EFFLUENTS

13. Mine wastes include liquids, solids, and exhaust ventilation gases. The liquids are mostly water from mine drainage and water used in drilling; it is discharged to ponds and allowed to evaporate. Leaching of mine tailings may be an additional source of liquid waste. Solid waste is primarily waste rock and low-grade ore. Ventilation of mines results in the discharge of large volumes of air containing radon and radon daughters.

14. The radionuclides contained in the ore include uranium, primarily ^{238}U, and the uranium daughter products, such as ^{230}Th and ^{226}Ra. In most cases little, if any, natural thorium (^{232}Th) is contained in the mill feeds. One tonne of ore containing 2 kg U_3O_8 has an activity of 566 μCi from each member of the ^{238}U decay chain, or a total of about 7900 μCi (126). About 14 per cent of the total activity contained in the ore fed to the mills is recovered in the uranium concentrate (14), while the average recovery of uranium is 91 per cent. In addition, the uranium concentrate contains about 5 per cent of the total thorium and 0.2 per cent of the radium contained in the ore (126). With little of the parent remaining, the short-lived daughters ^{234}Th (24.1 d), ^{234m}Pa (1.17 min) and ^{231}Th (25.5 h) are lost by decay from the mill wastes. The remaining 70 per cent of the activity originally present in the ore is largely undissolved and is discharged in the solid mill tailings.

15. Most of the radium in the ore is insoluble and remains in the tailing solids; a small portion, about 1 per cent or less, is dissolved. Radium in mill liquid effluents may vary from 250 to 500 pCi l^{-1} (14). Up to 50 per cent of the ^{230}Th in the ore may be dissolved in acid solutions and be contained in liquid effluents until the solution has been neutralized and the thorium precipitated. The radionuclides ^{223}R, ^{227}Th, ^{227}Ac, ^{210}Pb and ^{210}Po may also be present in the liquid effluents, but relatively little information is available on those nuclides because the ^{226}Ra level has been the most restrictive, making ^{226}Ra the most relevant nuclide of concern. At most mills, tailing ponds are of sufficient size for waste liquids to evaporate or seep into the ground. When space is not available, some liquid effluents are released into rivers, usually after treatment to reduce the radium content.

16. Releases of radioactive particles and radon gas into the air during milling operations may originate from the ore piles, the crushing, grinding and leaching operations, and the tailings piles. The release of radon activity from a dry tailings pile containing 560 pCi ^{226}Ra per gram is estimated to be 500 pCi m^{-2} s^{-1} (156). The release rate at any one location may vary, however, by an order of magnitude owing to the effects of such weather variables as wind speed, barometric pressure, atmospheric stability, rainfall, and snow cover (133). A pile saturated with water, as it may well be at an operating mill, may have its radon release rate reduced by a factor of 25 (156).

17. From a recent survey of milling practices at active uranium mills in the United States (126), the annual airborne radioactive releases from operating mills at two representative locations have been estimated. Table 1 gives the average and normalized (per MW(e) y) releases of various radionuclides from a mill processing 664 000 t of ore per year (151). The ore is assumed to be dusty and to contain 6 per cent moisture. Dust from ore crushing contains all radionuclides in equilibrium. However, yellowcake dust contributes to the uranium release and accounts for the lack of equilibrium in the total release. The sources of the airborne radioactivity in this estimate, in addition to the mill itself, include radon from the tailings pond and radon and particulates from the dry beach area. Since in this particular survey the maximum value of dry beach area was assumed for an active mill, namely the value that applies near the end of the assumed 20-y life of the mill, the releases from the dry beach area have been reduced by a factor of 2 to reflect more nearly average values for an operating mill. This correction mainly affects the amounts of radon released.

TABLE 1. ESTIMATED RADIOACTIVE AIRBORNE EFFLUENTS FROM MILLING OPERATIONS

Radio-nuclide	*Annual activity release from mill and active tailings area*[a] *(Ci)*	*Normalized release*[b] *(mCi per MW(e) y)*
^{238}U	0.09	0.01
^{234}U	0.09	0.01
^{234}Th	0.01	0.001
^{230}Th	0.01	0.001
^{226}Ra	0.009	0.001
^{210}Pb	0.009	0.001
^{210}Bi	0.009	0.001
^{210}Po	0.009	0.001
^{222}Rn	1 400	170

Source: Reference 126.

[a] Assuming an ore processing rate of 664 000 t y^{-1}.

[b] Assuming a uranium content of 0.2 per cent and an annual requirement of 160 t of uranium for a 1 GW(e) light-water reactor.

18. The tailings area remaining after the mill has ceased operation becomes a long-term source of environmental radioactive contamination due to wind or water erosion and to radon emanation. Stabilization programmes are generally conducted or planned so that erosion from tailing areas is alleviated (126). Radon-222 arises from decay of ^{226}Ra in secular equilibrium with long-lived ^{230}Th (80 000 y). A few per cent of uranium isotopes in the ore may also remain in the tailings material, constituting through the decay chain an even more long-lived source of ^{222}Rn emanation.

19. Emanation from the tailings arises primarily from radon produced in the surface layer. Of the total radon produced, up to about 20 per cent diffuses out of the crystallized structure of the mineral (28). The relatively short half-life of ^{222}Rn (3.8 d) results in the decay of much of the radon during gaseous diffusion towards the surface of the tailings pile. About 90 per cent of the radon released to the air originates in the top 1-2 m (124, 156), and the release rate is not changed by increasing the tailings depth much beyond about 3 m.

20. Covering the tailings pile with soil reduces the radon emanation rate by a factor of two for each metre of cover. Accordingly, about 6 m of earth cover is required to reduce the emanation to values observed naturally for soils (124).

B. LOCAL AND REGIONAL COLLECTIVE DOSE COMMITMENTS

21. Kristan *et al.* (78) have reported ^{226}Ra to be the most significant nuclide in water releases from mining and milling operations. When, as a result of releases, there is some increased radium concentration in fish, consumption of fish could be the dominant exposure pathway to individuals in the local population. Another possible pathway is via drinking water.

22. An inactive mill at Durango, Colorado (United States), has been studied extensively (137, 138). Before 1959, this mill discharged untreated effluents into the Animas River. Estimated ^{226}Ra release rates to the river were 0.2 mCi d^{-1} in dissolved form and 30 mCi d^{-1} in suspended solids (36). The radium concentration in the river was 12.6 pCi l^{-1} below the mill and 2.9 pCi l^{-1} 100 km downstream, compared to 0.3-0.6 pCi l^{-1} upstream. Concentration factors of 500-1000 for algae and aquatic insects, 3-4 for fish flesh and 100 for fish bone were reported (137). Because of the environmental contamination, waste treatment was introduced, which reduced the concentration of ^{226}Ra in the river to essentially the natural level (137). More recent surveys of uranium milling areas in the United States have not shown measurable water contamination beyond the plant sites (129, 130, 131).

23. Under the assumption that adequate waste treatment prevents seepage of process liquid effluents directly into rivers, the local collective dose commitments from milling operations are expected to depend on the airborne releases (table 1), which contribute to external exposure from deposited material and to internal exposure via the inhalation and ingestion pathways. The mills in the United States are typically located in areas not particularly suited for farming. The raising of beef cattle in the vicinity of the mills and the consumption of the beef may represent the only significant contribution to the intake by ingestion. Population in the milling areas is generally sparse and little growth can be expected. Based on a survey of five mills in the United States, Sears *et al.* (126) found that a population of about 5 10^4 within a radius of 80 km is representative. This corresponds to an average population density of less than 3 km^{-2}, with lower and non-uniform population densities very close to the mill. For areas at a greater distance, a population density of 25 km^{-2} would be representative. Variations in these figures, particularly the local population density, would be expected for specific mills in various other countries.

24. The collective dose from direct inhalation of radioactive materials released during operation can be assessed by considering the dispersion of the airborne particulates and radon with distance from the mill or tailings area. Annual average dispersion factors of about 10^{-6} s m^{-3} at 1 km have been found appropriate for the meteorology of typical mill locations (126), while for large area sources the value is smaller by a factor of about 4 (156). In this Annex, a value of 5 10^{-7} s m^{-3} at 1 km is assumed to apply for both the mill and the tailings area.

25. Over a limited range, the decrease of the concentration of activity in air with distance from the source appears to be approximately inversely proportional to the 1.5 power of the distance (126), but this behaviour cannot be easily verified by measurements since concentrations barely exceed natural levels at a few kilometres. Shearer and Sill reported measurements of radon concentrations 5, 1.5 and 0.15 times background at 0.8, 1.7 and 8.3 km respectively from inactive mill tailing areas (127). This corresponds to concentrations inversely proportional to the 1.5 power of the distance. Similar results were obtained by Breslin, who analysed various measurements of radon and particulate concentrations with distance from seven tailings areas (20). The relationship at much greater distances must be assumed to be only approximate, but it should serve to give indications of average dispersion over longer release periods.

26. Deposition of the airborne particulates reduces the area over which the integration of collective doses is carried out. For ground-level releases of particulate material, less than 30 per cent of the released material remains in the plume at 100 km and less than 20 per cent at 200 km (164). Integrating the collective doses to 500 km, therefore, will include virtually all the collective dose from airborne particulates released by the mill. Estimates of the collective doses from direct inhalation of particulates, calculated assuming the population distribution given in paragraph 23, are summarized in table 2.

27. The collective dose commitment from radon released during operations from the mill and the tailings area can be estimated by the same procedure used for the case of inhalation of particulates. For a radon release of 0.17 Ci per MW(e) y (table 1), and using a dispersion

TABLE 2. ESTIMATED COLLECTIVE DOSE COMMITMENT FOR AIRBORNE PARTICULATE RELEASES FROM MILLING OPERATIONS

		Collective dose commitment per unit activity released (man rad Ci^{-1})							*Normalized incomplete collective dose commitment (man rad per MW(e) y)*		
		Ingestion[a]			*Inhalation*			*External*			
Radio-nuclide	*Normalized activity released (Ci per MW(e) y)*	*Lung (whole)*	*Bone marrow*	*Bone. lining cells*	*Lung (whole)*	*Bone marrow*	*Bone lining cells*	*Whole body*	*Lung (whole)*	*Bone marrow*	*Bone lining cells*
^{238}U	1 10^{-5}	0.06	0.1	0.4	1.2	0.06	0.3		1 10^{-5}	2 10^{-6}	7 10^{-6}
^{230}Th	1 10^{-6}	0.06	0.03	0.4	19.0	12.0	180.0		2 10^{-5}	1 10^{-5}	2 10^{-4}
^{226}Ra	1 10^{-6}	0.04	0.1	1.0	3.6	0.09	0.6	10.0	1 10^{-5}	1 10^{-5}	1 10^{-5}
^{210}Pb[b]	1 10^{-6}	0.09	0.3	1.0	1.7	0.7	2.5		2 10^{-6}	1 10^{-6}	4 10^{-6}

[a] The values are the incomplete collective dose commitments due to deposition on the ground, incurred mainly through the ingestion pathway but also including a contribution from inhalation of resuspended materials. External exposure from deposited activity (^{226}Ra including its daughters) is considered separately.

[b] Including the contribution to dose from ^{210}Po daughter.

factor of 5 10^{-7} s m^{-3}, the normalized time-integrated concentration in air at 1 km is 0.024 pCi h l^{-1} per MW(e) y. Using a dose factor of 45 μrad per pCi h l^{-1} for radon in equilibrium with its daughters and an equilibrium factor $F = 0.6$ (see Annex B), the normalized dose commitment at 1 km is estimated to be 6 10^{-4} mrad per MW(e) y. At a few kilometres the equilibrium factor outdoors is probably smaller than 0.6 but would increase indoors because of the delay of the air circulated indoors. Therefore, not much error is introduced by assuming that the same value (0.6) applies over tens of kilometres. The local collective dose commitment within 100 km from the mill can be estimated to be

$$S^c = \int_1^{100} D_1 \left(\frac{\rho}{\rho_1}\right)^{-1.5} N_\delta \, 2\pi\rho \, d\rho = = 2\ 10^{-4} \text{ man rad per MW(e) y}$$

where D_1 is the dose commitment per unit electricity generated at the reference distance ρ_1 = 1 km, N_δ is the population density (3 km^{-2}), and ρ is the distance from the mill.

28. The regional collective dose commitment from 100 to 1500 km, obtained from similar integrations, but adjusted to account for radioactive decay and assuming a population density of 25 km^{-2} (para. 23), is 4.4. 10^{-3} man rad per MW(e) y. The sum of the local and regional components of the collective dose commitment is 5 10^{-3} man rad per MW(e) y. This value refers to the basal cells of segmented bronchioles; the corresponding collective dose commitments in the whole lung, gonads, bone marrow, and bone lining cells are listed in table 3.

29. An approximate estimate of the incomplete collective dose commitment due to the deposition of the released particulates by external exposure and the ingestion pathway, including inhalation of resuspended materials, may be obtained by assuming that the released nuclides, once deposited on the ground, behave similarly to their naturally occurring counterparts. Assuming that the deposited nuclides are distributed over a depth of 20 cm of soil and that the dose from all relevant exposure pathways is proportional to the activity concentration in that layer of soil, the incomplete collective dose commitment is given by

$$S^T_{i,j} = \frac{\dot{D}^*_{i,j}}{A^*_j} Q_j N_\delta \tau$$

where $S^T_{i,j}$ is the incomplete collective dose commitment in tissue i due to nuclide j, $\dot{D}^*_{i,j}$ is the dose rate in tissue i from the natural levels of nuclide j, A^*_j is the activity per unit surface area in the top 20-cm layer of the naturally occurring nuclide j, Q_j is the activity released of nuclide j (assumed to be equal to the deposited activity), N_δ is the population density, and τ is the assumed duration of the practice. It is assumed in this model that only decay removes the deposited nuclides from the soil.

30. Representative values for $\dot{D}^*_{i,j}$ and A^*_j can be found in Annex B. As most of the released particulate activity is deposited within a distance of 80 km (126), a value of 3 man km^{-2} is used for N_δ. Little is known about the effective residence time of these nuclides in soil. However, a conservative assessment of the incomplete collective dose commitment can be obtained by assuming that the availability of the nuclide decreases only by radioactive decay.

TABLE 3. COLLECTIVE DOSE COMMITMENTS FROM RADON RELEASED DURING ACTIVE MILL OPERATION[a]

(man rad per MW(e) y)

	Tissue		
Geographic region	*Segmental bronchioles*	*Lung (whole)*	*Gonads, bone marrow, bone lining cells*
Local (1-100 km)	2 10^{-4}	4 10^{-5}	3 10^{-7}
Regional (100-1 500 km)	4 10^{-3}	9 10^{-4}	7 10^{-6}

[a] Assuming a normalized release of 0.17 Ci per MW(e) y.

31. Table 2 presents the calculated incomplete collective dose commitments per unit activity released, the estimated amounts and the incomplete collective dose commitments per unit electrical energy generated. The incomplete commitments are integrated over the period of operation of the mills, and give some indication of the maximum future annual *per caput* doses to be expected from that operation, assuming the duration of the milling practice is 100 y. Reasonably assured reserves of uranium ore correspond to approximately 10^6 t of uranium (110), a value equivalent to some 6000 light-water reactor-years. At the presently installed capacity, the mining and milling practice could proceed for less than 100 y, unless further ore reserves become available.

32. The external exposure due to deposited ^{226}Ra will give an absorbed dose rate in air of the order of 30 rad h^{-1} per Ci m^{-2} for a plane source and 5 rad h^{-1} per Ci m^{-2} when the material is evenly distributed in the upper 20 cm of the soil, provided the activity per unit mass is considerably higher than in the soil material. Assuming a shielding factor of 0.32, the incomplete dose commitment will be in the range 4-25 man rad Ci^{-1} or (4-25) 10^{-6} man rad per MW(e) y. The average values are of the order of 10 man rad Ci^{-1} and 10^{-5} man rad per MW(e) y.

33. Estimating of the contribution to the collective dose commitment per MW(e) y made by radon release from the tailings pile remaining after the active life of the mill entails some sophisticated guesswork. Radon emanation could continue for millions of years because of the long half-life of the remaining ^{238}U precursor, but the continued presence of the pile on the surface for such a long period is unlikely. This Annex considers only the simpler but still uncertain case of an uncovered tailings pile undisturbed for times of the order of the half-life of ^{230}Th (80 000 y).

34. The uranium requirement of light-water reactors, 160 t per GW(e) y, is equivalent to 80 000 t of 0.2-per-cent ore, resulting in just slightly less than 80 000 t of solid tailings material. In the United States, more than 80 10^6 t of tailings (equivalent to 10^6 MW(e) y) occupied, 8 10^6 m^2 in 1970 (156), corresponding to 8 m^2 per MW(e) y.

35. For a radon emanation rate from the tailings piles of 500 pCi m^{-2} s^{-1} (156), and a dispersion factor of 5 10^{-7} s m^{-3}, the normalized concentration in air at 1 km is 2 10^{-6} pCi l^{-1} per MW(e) y, assuming that 1 MW(e) y corresponds to 8 m^2 of the tailings pile, as discussed in the previous paragraph. Over the mean life of ^{230}Th (1.44 X 80 000 y), the time-integrated radon concentration is 0.23 pCi y l^{-1} per MW(e) y. Using the dose factors and the dispersion assumptions discussed in previous paragraphs for the case of radon releases during operation of the mill, the collective dose commitments for radon release from the inactive tailings area can be assessed. The dose commitment to segmented bronchioles at 1 km is 50 mrad per MW(e) y. The local (1-100 km) collective dose commitment is estimated to be 20 man rad per MW(e) y, the regional (100-1500 km) value being about 400 man rad per MW(e) y. Both these values correspond to irradiation of the basal cells of segmental bronchioles.

36. An alternative estimate of the collective dose commitment due to radon release from the tailings of an inactive mill can be obtained by comparison with natural radon emanation from soil and the corresponding radon concentration in air. With the assumptions used above and assuming the natural radon exhalation rate to be 0.42 pCi m^{-2} s^{-1} and the corresponding radon concentration in air to be 0.1 pCi l^{-1} (see Annex B), the collective dose commitment is calculated to be

$$
\begin{aligned}
S^c &= (500\ \mathrm{pCi\ m^{-2} s^{-1}}) \left(\frac{0.1\ \mathrm{pCi\ l^{-1}}}{0.42\ \mathrm{pCi\ m^{-2} s^{-1}}}\right) \times \\
&\times (8\ \mathrm{m^2\ MW(e)^{-1}\ y^{-1}})(10^{-6}\ \mathrm{km^2\ m^{-2}}) \times \\
&\times (25\ \mathrm{man\ km^{-2}})(1.44)(80\ 10^3\ \mathrm{y})(0.6) \times \\
&\times (4.5\ 10^{-5}\ \mathrm{rad\ pCi^{-1} h^{-1} l})(8760\ \mathrm{h\ y^{-1}}) \\
&\approx 650\ \mathrm{man\ rad\ per\ MW(e)\ y}
\end{aligned}
$$

in reasonable agreement with the value assessed in the previous paragraph. The corresponding collective dose commitment would be 100 man rad per MW(e) y to the whole lung and 0.8 man rad per MW(e) y to the gonads, bone marrow and bone lining cells. The incomplete collective dose commitments (see Annex A), estimated for an assumed duration of the mining and milling practice of 100 y, are smaller than the above values by a factor of 10^3.

37. Table 4 summarizes the estimated values of the collective dose commitments due to radon release from the tailings area of inactive mills. These estimates are quite uncertain, and the true values will depend strongly on future practice. For example, if the fuel is reprocessed and if uranium is used in breeder reactors, the corresponding equivalent area of the tailings pile (8 m^2 per MW(e) y) will be much smaller and, therefore, the contribution to the collective dose commitment per MW(e) y made by milling will also be reduced.

TABLE 4. COLLECTIVE DOSE COMMITMENT TO TISSUES FROM RADON RELEASED FROM INACTIVE MILL TAILINGS

(man rad per MW(e) y)

Type of commitment	*Segmental bronchioles*	*Lung (whole)*	*Gonads, bone marrow, bone lining cells*
Complete	500	100	0.8
Incomplete	0.5	0.1	0.0008

Note: The commitment is for the local plus regional areas (radius 1-1500 km), assuming no soil cover. The values should be halved if a soil cover of 1 m is assumed.

II. URANIUM FUEL FABRICATION

38. The uranium ore concentrates produced at the mills are further processed and purified and usually enriched in the isotope ^{235}U before being converted to uranium dioxide or metal and fabricated into fuel elements. Natural uranium, which contains 0.7 per cent ^{235}U, can be utilized in heavy-water or graphite-moderated reactors. Light-water reactors require fuel enriched to 2-4 per cent ^{235}U.

39. Gaseous diffusion is the usual procedure for enrichment, but other methods can also be used. The U_3O_8 concentrate extracted from the ore must first be converted to the volatile compound, uranium hexafluoride (UF_6). The hydrofluor process consists of reduction, hydrofluorination and fluorination of the ore concentrates to produce crude UF_6, followed by fractional distillation to obtain the pure product. At the gaseous-diffusion installation, the UF_6 is pumped through a series of porous barriers which discriminate against the passage of the heavier isotope of uranium by a factor of 1.0043 at each stage. About 1700 barrier stages are required to produce an enrichment to 4 per cent (151).

40. In the final fuel fabrication step, UF_6 is converted again to an oxide or to the metal for use in fuel elements. For use as fuel in light-water reactors, the dioxide powder is pelletized, sintered and loaded into stainless steel or zircaloy tubing which is then capped and welded. Uranium-metal fuel for use in some gas-cooled reactors is clad in magnesium alloy (magnox). The fuel rods are assembled in arrays as fuel assemblies for installation in the reactor core.

A. EFFLUENTS

41. The radioactive materials discharged to the environment from fuel fabrication, conversion and enrichment installations are limited. Most of the uranium compounds are solid, and conventional air-cleaning equipment is used to remove particulates from airborne wastes. Liquid wastes are collected in settling tanks or ponds. Estimates of release amounts from typical installations, based on limited monitoring information, have been published in the United States (151). A summary of these estimates is given in table 5.

TABLE 5. ESTIMATED RADIOACTIVE RELEASES FROM FUEL FABRICATION OPERATIONS IN THE UNITED STATES

	Installation release rate ($Ci\ y^{-1}$)		*Equivalent installation size (GW(e))*	*Normalized release (10^{-6} Ci per MW(e) y)*		
	Airborne	*Liquid*		*Airborne*	*Liquid*	*Operation*
U	0.04	1.2	27.5	1.5	44.0	Conversion
	0.18	1.8	90.0	2.0	20.0	Enrichment
	0.005	0.5	26.0	0.2	20.0	Fabrication
^{226}Ra		0.09	27.5		3.4	Conversion
^{230}Th		0.04	27.5		1.5	Conversion
^{234}Th		0.26	26.0		10.0	Fabrication
^{234}Pa		0.26	26.0		10.0	Fabrication

Source: Reference 151.

42. Residual amounts of ^{226}Ra and ^{230}Th are removed from the uranium material in the conversion operation. Small amounts of these isotopes may be included in the liquid effluents from the conversion installations. The thorium daughters of uranium are separated at the mill, but ^{234}Th, with a half-life of 24 d, grows back to over 99 per cent secular equilibrium in 170 d; small amounts of ^{234}Th and its short-lived daughter ^{234}Pa are thus assumed present with uranium in liquid effluents from the fuel fabrication plant.

43. Estimates of the release of radioactive materials into the environment from enrichment and fuel fabrication installations in the United Kingdom are also available (106). The release of uranium to the atmosphere from the Capenhurst enrichment plant in the period 1972-1975 averaged 5 $kg\ y^{-1}$. Assuming 280 μCi of ^{238}U and of ^{234}U per kilogram of U_3O_8, the annual release is equivalent to 0.0028 Ci.

44. The alpha activity released annually in liquid effluents from the Capenhurst plant for the same period is estimated to be 0.09 Ci (106). The Springfields fuel fabrication plant in the United Kingdom of Great Britain and Northern Ireland discharged small amounts of liquid radioactive waste into the tidal river Ribble. Average alpha activity released annually in liquid effluents during 1972-1975 was 34 Ci (106).

B. LOCAL COLLECTIVE DOSE COMMITMENT

45. The liquid discharges from the Springfields fuel fabrication plant result in external exposure of individuals who frequent the river bank near the outfall. The annual collective dose due to the local population is estimated to be less than 0.1 man rad (100).

46. Estimates of collective dose commitments from airborne releases can be obtained in a manner similar to that used for releases from milling operations. The combined airborne release of uranium from all fuel fabrication operations, on the assumption that the values given in table 5 are representative, is about 4 10^{-6} Ci per MW(e) y. A population density of 25 km^{-2} has been assumed to apply at all relevant distances from the fuel fabrication installations. The estimated collective dose commitment assumes that the released uranium, once deposited on the ground, will remain available to enter the ingestion or the resuspension-inhalation pathway to man for the entire mean-life of ^{238}U. This assumption obviously leads to a considerable overestimation of the complete collective dose commitments. The values for the incomplete collective dose commitments (see Annex A) given in table 6 assume that fuel fabrication operations to serve reactors will be continued for a limited time, which for purposes of calculation is taken to be 500 y.

TABLE 6. ESTIMATED INCOMPLETE COLLECTIVE DOSE COMMITMENTS FOR URANIUM IN AIRBORNE EFFLUENTS FROM FUEL FABRICATION OPERATIONS[a]

Per unit activity released (man rad per Ci)						*Per unit electrical energy generated (man rad per MW(e) y)*		
Ingestion[b]			*Inhalation*			*Total*		
Lung	*Bone marrow*	*Bone lining cells*	*Lung*	*Bone marrow*	*Bone lining cells*	*Lung*	*Bone marrow*	*Bone lining cells*
0.5	0.8	4.0	2	0.09	0.4	1 10^{-5}	4 10^{-6}	2 10^{-5}

[a] Including conversion, enrichment and fabrication and assuming a normalized ^{238}U release of 4 10^{-6} Ci per MW(e) y.

[b] The values are collective doses due to deposition on the ground. They are mainly incurred through the ingestion pathway, but include also the contribution from inhalation of resuspended materials.

III. REACTOR OPERATION

47. The common reactor types include the pressurized-water moderated and cooled reactor (PWR), the boiling-water moderated and cooled reactor (BWR), the gas-cooled, graphite-moderated reactor (GCR), the water-cooled, graphite-moderated reactor (LWGR), the heavy-water moderated reactor (HWR), which may be cooled by heavy water, light water or gas, and the fast breeder reactor (FBR).

48. Table 7 gives the number of operational reactors of each type and the electric generating capacity for each country at midyear in 1976 (74). The total installed nuclear generating capacity in the world in 1976 was 79.8 GW(e) from 188 reactors in 19 countries.

A. EFFLUENTS

49. During the operation of a nuclear reactor, radioactive fission and activation products are produced. These radioactive materials are for the most part retained within the fuel elements. Most of the radionuclides which diffuse into or are formed within the coolant are removed by the gaseous and liquid waste processing systems. Low-level releases, which occur during normal operation, are controlled and monitored.

50. The type and quantity of radioactive materials released from reactors depends on the reactor type and on the specific waste processing systems utilized. Radionuclides may reach the environment through either the gaseous or liquid effluent streams. In the airborne effluents are found fission noble gases (krypton and xenon isotopes), activation gases (^{41}Ar, ^{14}C, ^{16}N, ^{35}S), tritium, radioactive halogens and particulates. In the liquid effluents are tritium, fission products and activated corrosion products.

1. Fission noble gases

51. Several isotopes of the noble gases krypton and xenon occur by fission in the nuclear fuel. The total volume of xenon and krypton generated is about 25 cm^3 NTP per thermal megawatt-day. The fraction that escapes from the fuel pellets during normal operation generates gas pressure in the fuel can. The appearance of noble gas activity in effluent streams results mainly from fuel-cladding failure. Only insignificant amounts are formed in the coolant by fission of trace uranium on external fuel-element surfaces.

52. In PWRs the primary coolant is maintained in a sealed loop, which is opened only infrequently. Therefore, much of the short-lived noble gas activity decays before venting. Short-lived noble gases originate mainly from water that has leaked from the primary coolant system.

53. A side stream of the primary coolant in PWRs is continually diverted into a subsystem (primary coolant letdown) for control of chemical content and for coolant purification. The gaseous waste effluents from this system are compressed into tanks and held for a period of 30-120 d for decay of short-lived isotopes. Other gaseous effluent streams in PWRs originate from the main condenser exhaust, secondary coolant blowdown, reactor building ventilation including containment purges (~4 times a year), and turbine and auxiliary building ventilation (166).

54. In table 8, which lists the reported noble gas releases from PWRs, it can be seen that the normalized release amounts span a considerable range. Leakage is an important and variable source term. The total activity released divided by the net electrical energy generated for all PWRs for which effluent data are available varied from 10 to 20 Ci per MW(e) y during 1970-1974. The composition of the release from PWRs is primarily longer-lived ^{133}Xe (5.3 d), but some shorter-lived noble gas isotopes are also present. Krypton-85 (10.7 y) may provide a few per cent to the PWR release activity composition. The activity of specific noble gas isotopes in the releases reported for PWRs in the Federal Republic of Germany and in the United States during 1974 are given in table 9.

55. In BWRs, continuous removal of non-condensable gases in the steam flow occurs via the main condenser air-ejector system. The noble gases enter the gaseous waste stream primarily at this point. Secondary

TABLE 7. WORLD NUCLEAR ELECTRIC GENERATING CAPACITY IN 1976

Net values in MW(e) and number of reactors in parentheses

	Reactor type						
Country	*PWR*	*BWR*	*GCR*	*HWR*	*LWGR*	*FBR*	*Total*
United States	24 398 (34)	15 208 (23)	330 (1)				39 936 (58)
United Kingdom			6 660 (31)	92 (1)		244 (2)	6 996 (34)
Japan	2 879 (5)	3 264 (7)	154 (1)				6 297 (13)
Germany, Federal Republic of	4 033 (5)	1 148 (4)	14 (1)	52 (1)			5 247 (11)
Union of Soviet Socialist Republics	2 094 (6)	94 (5)			2 882 (11)	146 (2)	5 216 (24)
Sweden	822 (1)	2 362 (4)					3 184 (5)
France	270 (1)		2 133 (7)	70 (1)		250 (1)	2 723 (10)
Canada				2 540 (7)			2 540 (7)
Belgium	1 663 (4)						1 663 (4)
Spain	153 (1)	440 (1)	480 (1)				1 073 (3)
Switzerland	700 (2)	306 (1)					1 006 (3)
German Democratic Republic	879 (3)						879 (3)
Bulgaria	837 (2)						837 (2)
India		396 (2)		207 (1)			603 (3)
Italy	242 (1)	150 (1)	150 (1)				542 (3)
Netherlands	447 (1)	52 (1)					499 (2)
Argentina				319 (1)			319 (1)
Pakistan				125 (1)			125 (1)
Czechoslovakia				110 (1)			110 (1)
Total	39 416 (66)	23 420 (49)	9 921 (43)	3 423 (13)	2 882 (11)	640 (5)	79 794 (188)

Source: Reference 66.

pathways include the turbine gland-seal system, the condenser mechanical vacuum pump, and any leakage of process fluids to ventilated building spaces (165).

56. Table 8 also lists the reported noble gas released activities from BWRs. The normalized releases vary from less than 100 to more than 10 000 Ci per MW(e) y. The value representing the overall operating experience has declined from about 1500 Ci per MW(e) y in 1970 to 1000 Ci per MW(e) y in 1974. The release composition of noble gas isotopes from BWRs varies somewhat, dependent on the specific hold-up time. The hold-up times range from 20 to 30 min for older BWRs to 2 h or more for more recently designed hold-up systems. The reported release compositions for BWRs in the Federal Republic of Germany and in the United Sates during 1974 are given in table 9. Important components of the releases are ^{135}Xe (9.2 h), ^{133}Xe (5.3 d), ^{138}Xe (17 m), ^{88}Kr (2.8 h), ^{87}Kr (76 m) and ^{85m}Kr (4.4 h). The specific release activities are not necessarily typical for any one reactor, but the total normalized release is similar to the overall average from table 8.

57. In GCRs, fission noble gas release is insignificant. Gas-cooled reactors include the Magnox type, the advanced gas-cooled reactor (AGR) and the high-temperature gas-cooled reactor (HTGR). The Magnox type utilizes metallic uranium fuel and CO_2 coolant. Because of the magnesium-alloy cladding fuel temperatures must be kept at less than 650°C, at which temperature release of noble gases from the fuel material is negligible.

TABLE 8. NOBLE GASES DISCHARGED IN AIRBORNE EFFLUENTS FROM REACTORS IN VARIOUS COUNTRIES, 1970-1974

Reactor	*Start-up year*	*Net electrical power (MW(e))*	*Release (hCi)*					*Normalized release (Ci per MW(e) y)*				
			1970	*1971*	*1972*	*1973*	*1974*	*1970*	*1971*	*1972*	*1973*	*1974*
PWR												
Belgium												
BR-3, Mol	1962	10			0.25					190		
France												
SENA, Chooz	1967	270	0.003	4.5	31.3	19.9		0.02	20	130	82	
Germany, Fed. Rep. of												
Obrigheim	1968	328	7.7	1.46	3.20	2.93	13.5	27	5.7	12		
Stade	1972	630			2.45	2.61	0.89			6.5		
Biblis	1974	1 147					0.06					
Italy												
Trino, Vercellese	1964	247	0.019	0.59	1.03			0.13	3.8	4.5		
Japan												
Mihama 1	1970	340	0.9	1.4	0.62	0.51	0.07	11	5.6	4.9	5.4	2.8
Mihama 2	1972	500			0.26	0.34	0.34			1.1	1.2	1.1
Takahama 1	1974	826					0.07					0.27
Netherlands												
Borssele	1973	447				0.31	5.83				4.0	18
United States												
Yankee Rowe	1961	175	0.017	0.013	0.018	0.035	0.040	0.12	0.08	0.25	0.29	0.38
Indian Pt. 1	1962	265	1.7	0.36	0.54	0.12	0.61	39	4.1	3.8	–	4.5
San Onofre	1968	430	4.2	7.67	19.1	11.0	1.78	12	21	59	42	5.0
Connecticut Yankee	1968	575	0.7	3.25	0.65	0.032	0.0074	1.7	6.8	1.3	0.12	0.015
R. E. Ginna	1970	420	10.0	31.8	11.8	0.58	0.76	38	100	43	1.5	3.2
Point Beach, 1,2	1970/72	2 × 497		0.84	2.81	5.75	9.74		2.1	7.9	9.9	13
H. B. Robinson	1971	700		0.018	0.26	3.1	2.31		0.061	0.51	7.2	4.2
Palisades	1971	700			0.51	0.45	0.00003			2.3	1.7	0.0038
Maine Yankee	1972	790			0.002	0.16	6.36			0.04	0.41	16
Surry, 1,2	1972/73	2 × 788			0.00001	0.87	55.0			0.0003	1.0	82
Turkey Pt. 3,4	1972/73	2 × 693				0.53	4.66				0.99	5.2
Indian Pt. 2	1973	873				0.015	5.58				0.38	15
Ft. Calhoun	1973	457				0.066	0.30				0.96	1.1
Prairie Island, 1,2	1973/74	2 × 530				0.008	0.36				3.6	2.2
Oconee 1,2,3	1973/74	3 × 886				9.3	19.4				35	31
Zion 1,2	1973	2 × 1 050				0.004	2.99				0.052	5.6
Arkansas 1	1974	820					0.20					3.0
Kewaunee	1974	520					3.35					18
Three Mile Island	1974	810					0.92					3.8
		Total	25.2	51.9	74.8	58.6	135.1					
Electrical energy generated (MW(e) y)			1 906	3 124	3 960	6 083	9 045					
Overall normalized release (Ci per MW(e) y)			13.2	16.6	18.9	9.6	14.9					
BWR												
Germany, Fed. Rep. of												
Kahl												

Overall normalized release (Ci per MW(e) y)			13.2									
BWR												
Germany, Fed. Rep. of												
Kahl	1961	15	2.90	2.10	–	0.48	0.99	220	160	–	37	95
Gundremmingen	1966	237	7.35	6.50	11.0	23.7	4.15	35	29	53	120	19
Lingen	1968	174	110	8.70	5.1	2.6	9.5	960	75	84	18	180
Würgassen	1972	640			0.59	0.56	0.05			9	2.4	0.9
Italy												
Garigliano	1964	152	275	640	290			3 300	4 800	6 000		
Japan												
Tsuruga	1969	331	130	42	4.9	5.2	5.6	470	170	19	18	33
Fukushima 1,2	1970/74	460/784	1.2	70	97	4.9	14	160	230	330	22	29
Netherlands												
Dodewaard	1968	52	2.41	2.76	8.32	6.70	4.16	57	60	220	160	130
United States												
Dresden 1	1960	200	900	753	877	840	98.4	5 700	9 500	7 000	13 000	2 500
Big Rock Pt.	1962	70	280	284	338	230	188	6 100	7 200	5 900	4 800	4 900
Humboldt Bay	1963	69	540	514	430	350	572	11 000	14 000	10 000	7 300	13 000
Lacrosse	1969	53	0.95	0.53	30.6	91	49.0	64	21	1 100	4 000	1 300
Oyster Creek	1969	640	110	516	866	810	279	270	1 200	1 700	2 000	660
Nine Mile Pt.	1969	625	9.5	253	517	872	558	44	770	1 400	2 200	1 500
Dresden 2,3	1970/71	2 X 800	250	580	429	880	627	1 800	1 300	460	890	830
Millstone Pt. 1	1970	652		276	726	79	912		650	2 000	370	2 200
Monticello	1971	545		75.8	751	870	1 490		450	1 900	2 300	4 500
Quad Cities, 1,2	1972	2 X 800			132	900	950			300	780	1 000
Pilgrim 1	1972	655			18.1	230	546			200	490	2 400
Vermont Yankee	1972	514			55.2	180	63.9			1 100	880	220
Peach Bottom 2,3	1973/74	2 X 1 000				0.004	0.2					0.4
Browns Ferry	1973	1 050					64					150
Cooper Station	1974	760					1.6					7.6
		Total	2 619	4 024	5 587	6 376	6 438					
Electrical energy generated (MW(e) y)			1 790	3 121	4 369	5 552	6 166					
Overall normalized release (Ci per MW(e) y)			1 460	1 290	1 280	1 150	1 040					
GCR												
France	1963/	70										
EDF 1,2,3 Chinon	1965/	200	8.09	4.23	11.52	2.8		19.6	10.9	23.2	8.7	
	1966	480										
SL 1,2 St. Laurent	1969/ 1970	480 505	0.31	3.43	3.86	5.0		19.3	9.5	5.9	7.0	
Bugey	1972	540			0.84	3.1				6.5	10.6	
Germany, Fed. Rep. of												
AVR-Jülich	1968	13	0.031	0.027	0.028	0.021	0.010	3	2.4	2.7		
Italy												
Latina	1963	153	2.50	2.47	3.66			18.4	11.7	26.6		
Japan												
Tokai	1965	166	4.9	4.8	5.9	6.0	6.3	46.7	42.1	52.7	52.6	55.8
		Total	15.8	14.9	25.8	16.9	6.3					
Electrical energy generated (MW(e) y)			880	1 085	1 546	1 444	113					
Overall normalized release (Ci per MW(e) y)			18.0	13.7	16.7	11.7	55.8					

TABLE 8 *(continued)*

Reactor	Start-up year	Net electrical power (MW(e))	Release (kCi)					Normalized release (Ci per MW(e) y)				
			1970	1971	1972	1973	1974	1970	1971	1972	1973	1974
HWR												
Argentina												
Atucha	1974	320					1.8					17
Canada												
Douglas Pt.	1966	220	161					1 500				
Pickering 1,2,3,4	1971/72/73	4 X 508		58.1	40.0	31.0	3.1		170	57	19	2
France												
EL-4-Monts d'Arrée	1967	70	0.072	53.8	144.5	130			2 700	2 500	2 500	
Germany, Fed. Rep. of												
MZFR-Karlsruhe	1965	51		0.53	0.96	0.22	0.95		14	19	19	23
		Total	161	112	186	161	5.9					
Electrical energy generated (MW(e) y)			107	402	808	1 694	1 770					
Overall normalized release (Ci per MW(e) y)			1 500	280	230	95	3					

Sources: References 1, 26, 27, 43, 44, 97-99, 120, 141, 143, 144, 161, 163.

TABLE 9. COMPOSITION OF NOBLE GASES DISCHARGED IN AIRBORNE EFFLUENTS FROM REACTORS

Reactor	Electrical energy generated in 1974 (MW(e) y)	Amount released (Ci)												
		^{133}Xe	^{135}Xe	^{138}Xe	^{85}Kr	^{131m}Xe	^{88}Kr	^{135m}Xe	^{85m}Kr	^{133m}Xe	^{87}Kr	^{41}Ar	^{37}Ar	Total
PWR														
Germany, Fed. Rep. of														
Obrigheim	293.4	12 100	1 320									35		
Stade	608.2	711	156		23									
United States														
Yankee Rowe	103.9	16.4	1.2	0.0005	1.5	0.02	0.03	0.0006	0.06	0.02	0.02	0.9	0.5	
San Onofre	353.9	1 420	9.7		63.9	105			0.9	177				
Connecticut Yankee	502.3	2.3	0.4		4.0		0.3	0.3			0.2			
R. E. Ginna	239.7	630	110		8.9				7.1			1.1		
Pt. Beach 1,2	764.8	6 040	1 300	321	23.1	394	542	322	374	133	251	32.0		
H. B. Robinson	547.9	2 110	43.1			0.8			2.8			3.2		
Palisades	8.9	0.02	0.01			0.008			0.0002	0.00003				
Maine Yankee	411.0	5 460	23.7		532	88.6				3.3				
Surry 1,2	673.5	54 600	223		8.1		19.3		19.8	2.6	14.9	15.0		
Turkey Pt. 3,4	901.8	3 980	172	15.8	291	38.0	14.5	51.1	17.5	22.2	4.7	52.1		
Ft. Calhoun	274.0	0.03	1.2		0.5									

Prairie Island	160.0	357	0.1		0.0005		0.03	0.03	0.001	0.2		1.1		
Oconee 1,2,3	632.4	18 600	527		0.7	16.2	37.5		74.5	37.2	1.6	87.4		
Zion 1,2	536.5	2 980	6.3											
Arkansas 1	65.1	0.02			0.06	0.02			0.03	2.5	0.0001	0.0001		
Kawaunee	182.6	2 690	128					0.00001	2.2	11.1		2.7		
Three Mile Island	239.7	891	12.6			0.7	0.003	0.0002	0.5	6.7		5.8		
	Total	112 600	4 035	336.8	956.8	643.3	613.7	373.4	499.4	395.8	272.4	236.3	0.5	
Corresponding electrical energy (MW(e) y)		7 500	7 435	1 771	5 690	4 029	3 978	2 855	4 874	4 498	3 644	4 804	104	
Normalized release (Ci per MW(e) y)		15.0	0.54	0.19	0.17	0.16	0.15	0.13	0.10	0.09	0.07	0.05	< 0.01	17
Proportion of total (%)		90	3.2	1.1	1.0	1.0	0.9	0.8	0.6	0.5	0.4	0.03	< 0.1	100

		Amount released (kCi)												
		^{135}Xe	*^{88}Kr*	*^{133}Xe*	*^{138}Xe*	*^{87}Kr*	*^{83m}Kr*	*^{85m}Kr*	*^{135m}Xe*	*^{85}Kr*	*^{137}Xe*	*^{133m}Xe*	*Other*	
B W R														
Germany, Fed. Rep. of														
Kahl	10.4	0.6	0.06		0.02	0.04		0.2				0.02	0.04	
Gundremmingen	219.2	0.2	0.3	1.9	0.4	0.1	0.04	0.5	0.1	0.04	0.3		0.3	
Lingen	52.9	1.7	1.4	0.6	0.7	1.0	0.2	0.4	0.6	0.04	1.0	0.03	0.7	
Würgassen	55.7	0.02	0.004	0.001	0.002	0.007		0.005	0.001				0.02	
United States														
Dresden 1	40.0	10	5.7	2.1	48	11		1.6	20					
Big Rock Pt.	38.8		17.5		54.8	30.2		10.2	8.2	0.4	3.8	0.7	3.7	
Humboldt Bay	43.4	107	76.4	36.2	169	75	14.4	21.9	65.6		4.7	1.2	0.8	
Lacrosse	37.7	6.6	3.8	1.7	18.2	4.8		1.5	8.9	0.005	3.3	2.9	0.2	
Oyster Creek	422.4	73.4	50.7	22.6	55.1	47.3		16.1	13.9		0.03	0.2	0.006	
Nine Mile Pt.	376.7	182	133	59.3	61.5	78.7		43						
Dresden 2,3	753.4	185	95.1	83.8	67.6	111		38.3	46.5					
Millstone Pt.	411.0	257	142	255	66.5	89.9		51.6						
Monticello	331.1	533	200	216	223	210		98.4		9.8				
Quad Cities 1,2	924.7	270	175	193	61	127	125	61	25.2	6.6				
Pilgrim 1	228.3	153	90.8	81.4	63	71.2		36.3						
Vermont Yankee	285.4	34.8	0.8	22.2	3.2	1.2		1.7						
Browns Ferry	593.6	0.5	0.8	4.5	5.8	1.4		0.4	6.3	45.6			1.5	
Cooper Station	205.5	0.1	0.001	0.01		0.2	0.02	0.04	0.2		0.05			
	Total	1 814.9	993.4	980.3	897.8	860.0	139.6	383.1	195.5	62.5	13.2	5.0	6.4	
Corresponding electrical energy (MW(e) y)		4 991	5 030	4 981	4 825	5 030	1 446	5 030	3 388	2 403	1 020	606	1 474	
Normalized release (Ci per MW(e) y)		360	200	200	190	170	100	80	60	30	10	8	4	1 400
Proportion of total (%)		26	14	14	13	12	7	5	4	2	0.9	0.6	0.3	100

Sources: References 55, 163.

58. The AGR utilizes slightly enriched UO_2 as fuel and CO_2 as coolant. The cladding is stainless steel. Somewhat higher operating temperatures are reached, about 1300°C, but noble gas release is still expected to be quite low. Commercial AGRs have come into operation only recently.

59. The helium-cooled HTGR utilizes graphite-coated, highly enriched UO_2 or UC_2 as fuel. The prototype HTGRs thus far constructed, AVR (Federal Republic of Germany), Dragon (United Kingdom) and Peach Bottom (United States), have utilized charcoal filters to retain the noble gases which reach the primary coolant for long periods. The release composition is thus primarily ^{133}Xe and ^{85}Kr. Noble gas release reported from Peach Bottom averaged about 5 Ci per MW(e) y for 1968-1973 (141, 161). For the AVR reactor, the noble gas release has been 2-3 Ci per MW(e) y for 1969-1972 (27). Larger HTGRs are expected to release noble gases at the rate of 1-3 Ci per MW(e) y (49).

60. Table 8 also gives a limited amount of data on the total activity of noble gases released from GCRs and HWRs. In some cases, measurements at GCRs did not distinguish fission noble gases from ^{41}Ar. Therefore, in reports of release data from these reactors, the amounts of fission noble gases are upper limits.

2. Activation gases

61. In contrast to the noble gas fission products released in BWR and PWR operation, several activation gases are formed in GCR operation. Direct activation of the CO_2 coolant gives rise to ^{16}N by the (n, p) reaction with ^{16}O. Argon-41 arises from the (n, γ) reaction with ^{40}Ar in the coolant and also in shield-cooling air, where this is used.

62. The amounts of ^{41}Ar released depend on the reactor characteristics. Release rates of ^{41}Ar from GCRs in the United Kingdom are not measured routinely, but for a Magnox reactor, the rate has been estimated to be approximately 20 Ci h^{-1} from shield cooling and an additional 10 Ci d^{-1} from CO_2 leakage (38). A number of recent measurements at three reactors in the United Kingdom indicate ^{41}Ar releases on the order of 100-300 Ci per MW(e) y (115). Estimated discharges for an AGR due to a coolant leakage of 1 per cent per day are about 3 Ci per MW(e) y for ^{41}Ar and 2 Ci per MW(e) y for ^{16}N (113).

63. Another airborne effluent from GCRs is ^{35}S, arising from (n, γ) reactions with ^{34}S and (n, p) reactions with ^{35}Cl. The impurity levels of sulphur and chlorine in moderator graphite are approximately 50 and 2 ppm, respectively (113). Release of ^{35}S from the Oldbury and Wylfa (Magnox) reactors in the United Kingdom averaged 0.002 Ci per MW(e) y during 1972-1974 (106).

64. Nitrogen-16 (7.3 s) is also formed in the coolant in light-water reactor cores. In BWRs a portion of the ^{16}N is transported with the steam before decay to the less-shielded turbines. The photons (6.1 and 7.1 MeV) from ^{16}N decay in the turbines are a potentially significant component of the gamma radiation field near the turbine building (86, 87).

3. Tritium

65. Tritium occurs by ternary fission in nuclear fuel and also by neutron activation reactions with lithium and boron isotopes dissolved in, or in contact with, the primary coolant and with naturally occurring deuterium in the primary coolant. Erdman (31) has estimated the normalized tritium generation rate in LWRs from ternary fission to be about 15 Ci per MW(e) y, based on a recent measurement of 0.85 10^{-4} tritons per thermal fission in ^{235}U (34), and assuming 2 10^{-4} tritons per thermal fission in ^{239}Pu and ^{238}U (31). The estimate assumes 55 per cent fissions in ^{235}U, 41 per cent in ^{239}Pu and 4 per cent in ^{238}U. Other estimates indicate about 23 Ci per MW(e) y (107). An average assumed for use in this Annex is 20 Ci per MW(e) y. The generation rate estimate for a fast breeder reactor is also 20 Ci per MW(e) y (31).

66. Estimated generation rates and appearance of tritium in effluent streams of reactors are summarized in table 10. Assessments of tritium activity in the coolant have generally assumed that 1 per cent of the tritium generated in the fuel is transferred to the coolant through pinhole defects in the fuel element cladding. In earlier PWRs, which utilized stainless steel cladding, as much as

TABLE 10. ESTIMATED RATES OF GENERATION OF TRITIUM AND OF ITS APPEARANCE IN EFFLUENT STREAMS OF DIFFERENT TYPES OF REACTORS

(Ci per MW(e) y)

	BWR		*PWR*		*GCR*		*HWR*	
Source	*Generation*	*Effluent stream*	*Generation*	*Effluent stream*	*Generation*	*Effluent stream*	*Generation*	*Effluent stream*
Fission	20	0.2	20	0.2	20	0.2	20	0.2
Soluble boron shim			0.5	0.5				
Deuterium activation	0.01	0.01	0.001	0.001			600	20[a]
Lithium activation			0.02	0.02	0.5	0.1		
Boron activation (control rods)	9							
Total	29	0.2	21	1	21	0.3	620	20

Sources: References 39, 76, 128, 136.

[a] Depending on the irradiation time and on the net leakage of heavy water.

80 per cent of the fission-produced tritium could permeate the cladding. With the zircaloy cladding in current usage, 1 per cent or less of the fission-produced tritium is expected to appear in the effluent streams (85). Although tritium is formed within the control rods in BWRs, it has not been shown to diffuse through the B_4C matrix of the control rods (136).

67. Reports on PWR operation (tables 11 and 12) show the average normalized release of tritium to be about 2 Ci per MW(e) y in liquid effluents and 0.2 Ci per MW(e) y in airborne effluents. For BWRs, the averages are 0.1 and 0.05 Ci per MW(e) y in liquid and airborne effluents respectively.

68. In GCRs, tritium is produced by activation of lithium impurities in the graphite moderator as well as by ternary fission. Use of helium as coolant in HTGRs results in tritium generation by the (n, p) reaction with ^{3}He. Estimates of the normalized generation rates are 0.5 Ci per MW(e) y from lithium activation (39) and 3-6 Ci per MW(e) y from helium activation (136).

TABLE 11. TRITIUM DISCHARGED IN AIRBORNE EFFLUENTS FROM REACTORS IN VARIOUS COUNTRIES, 1973-1974

	Release (Ci)		*Normalized release (Ci per MW(e) y)*	
Reactor	*1973*	*1974*	*1973*	*1974*
P W R				
Germany, Fed. Rep. of				
Obrigheim	20.2	11.5	0.067	0.039
Stade		11.1		0.018
United States				
Yankee Rowe	8.4	3.8	0.070	0.037
Indian Pt. 1	25.4	0.32		0.0024
San Onofre	268.9	91.4	1.02	0.26
Connecticut Yankee	50.6	0.012	0.18	0.000023
R. E. Ginna	1.1	0.37	0.0028	0.0015
Pt. Beach 1,2	25.5	42.8	0.039	0.056
H. B. Robinson	2.5	51.5	0.0058	0.094
Palisades	0.18	0	0.00066	0
Maine Yankee	1.89	7.2	0.0049	0.017
Surry 1,2	42.4	60.4	0.051	0.090
Turkey Pt. 2,3	4.1	9.2	0.0076	0.010
Indian Pt .2	2.0	19.9	0.050	0.053
Ft. Calhoun	0.33	0.75	0.0048	0.0027
Prairie Island		3.9		0.024
Oconee 1,2	13.1	878	0.049	1.39
Zion 1,2		180		0.34
Arkansas 1		0.030		0.00046
Kewaunee		109		0.60
Three Mile Island		12.7		0.053
Total	466	1 494		
Electrical energy (MW(e) y)	5 393	8 014		
Overall normalized release (Ci per MW(e) y)	0.09	0.19		
B W R				
United States				
Big Rock Pt.	77.1	38.7	1.61	1.00
Humboldt Bay	1.9	1.7	0.040	0.040
Lacrosse	50.6	18.3	2.24	0.49
Oyster Creek	0.32	0.42	0.00078	0.00098
Nine Mile Pt.	26.8	15.8	0.067	0.042
Dresden 2,3	10.0	114	0.010	0.15
Millstone Pt.	1.7	7.9	0.0079	0.019
Monticello				
Quad Cities 1,2	34.0	29.0	0.030	0.031
Pilgrim 1	14.0	8.0	0.030	0.035
Vermont Yankee	1.0	2.2	0.0049	0.0078
Peach Bottom 2,3		5.6		0.0094
Browns Ferry		0.65		0.0015
Cooper Station		0.016		0.000077
Total	217	242		
Electrical energy (MW(e) y)	4 340	5 094		
Overall normalized release (Ci per MW(e) y)	0.050	0.048		

TABLE 11 *(continued)*

Reactor	*Release (Ci)*		*Normalized release (Ci per MW(e) y)*	
	1973	*1974*	*1973*	*1974*
H W R				
Argentina				
Atucha		217		2.0
Canada				
Pickering	36 500	24 800	22.4	15.4
Total	36 500	25 017		
Electrical energy (MW(e) y)	1 630	1 723		
Overall normalized release (Ci per MW(e) y)	22.4	14.5		
G C R				
United Kingdom				
Oldbury	30	12	0.10	0.037
Wylfa	200		0.67	
Total	230	12		
Electrical energy (MW(e) y)	588	321		
Overall normalized release (Ci per MW(e) y)	0.39	0.04		

Sources: References 26, 44, 106, 161, 163, 171.

69. The release rate of tritium from GCRs varies with reactor type. Penetration of the Magnox cladding by tritium produced in fuel can be assumed to be about 1 per cent. For HTGRs an estimated 30-90 per cent of the tritium originating in the fuel and graphite moderator may reach the coolant (39, 136). The tritium is removed in coolant-gas purification, but a 4 per cent diffusion loss through the condenser walls is anticipated, or about 3 Ci per MW(e) y (39). Based on the experience with GCRs, primarily for the Magnox reactors in the United Kingdom, tritium release in liquid effluents (table 12) is about 0.2 Ci per MW(e) y, and in airborne effluents it ranges from 0.04 to 0.4 Ci per MW(e) y, on the basis of the limited information available (table 11).

70. The amount of tritium generated in HWR fuel by ternary fission is approximately the same as in LWRs, but activation of deuterium in the D_2O coolant and moderator provides much more. Kouts estimates the tritium production rate in the coolant to be 600 Ci per MW(e) y (76). Tritium concentration in the primary coolant and moderator of a HWR depends upon the irradiation time of the heavy water and the reactor size. The D_2O leakage from HWRs is kept as small as possible for economical and radiological reasons. Leakage from the MZFR reactor in the Federal Republic of Germany averages about 1.5 l d^{-1}, about 0.5 per cent of the total D_2O amount per year (39). For larger reactors, annual losses of from 1 per cent (Atucha reactor in Argentina and Pickering reactors in Canada) to 2-3 per cent can be anticipated. For the optimal loss of 0.5 per cent per year, the normalized tritium release rate from a HWR ranges from 3 Ci per MW(e) y in the first year of operation to about 20 Ci per MW(e) y in the tenth. Based on the latter value as representative of the reactor life, the normalized tritium release is estimated to be 16 Ci per MW(e) y in airborne effluents and 4 Ci per MW(e) y in liquid effluents (39). For larger leakage rates, on the order of 1 per cent per year, these values would be attained after a few years of reactor operation. Information on actual releases of radioactivity from HWRs is limited; however, the data reported for the Pickering reactors in Canada (tables 11 and 12) roughly agree with these estimates (171). For the Atucha reactor in Argentina, normalized tritium release in liquid effluents was 0.8 Ci per MW(e) y during its initial year of operation (1974) and 3 Ci per MW(e) y in 1975 (26). Tritium in airborne effluents from Atucha was 2.0 and 3.8 Ci per MW(e) y in 1974 and 1975, respectively (26).

4. Carbon-14

71. Very little attention has been given to ^{14}C production in and release from reactors. The dose rates associated with the releases are very low; however, with its long half-life (5730 y), ^{14}C can make a significant contribution to the collective dose commitment.

72. Carbon-14 is produced in LWRs and HWRs by (n, α) reactions with ^{17}O present in the oxide fuel and in the moderator, by (n, p) reactions with ^{14}N present in the fuel as impurities, and by ternary fission. Ternary fission production per unit electrical energy generated is practically independent of reactor design, while the normalized production of ^{14}C by the other reactions depends on the enrichment of the fuel, the relative masses of the fuel and moderator, the concentration of nitrogen impurities in the fuel and the temperature of the fuel and moderator.

73. In 1974, Bonka estimated the normalized ^{14}C production in light water reactors to be about 0.03 Ci per MW(e) y (18). Magno, also in 1974, estimated the production to be about 0.05 Ci per MW(e) y (90). Hayes *et al.* (45) estimated the production to be 0.02 Ci per MW(e) y for PWRs and 0.04 Ci per MW(e) y for BWRs. Recent estimates by Davis (29) gives values of 0.02 Ci per MW(e) y in PWRs and BWRs. A study carried out for Swedish power reactors (84) estimated the values to be 0.014 Ci per MW(e) y for PWRs and 0.024 Ci per MW(e) y for BWRs.

74. Assuming, as in the assessments mentioned above, that the average thermal neutron flux density is the same in the fuel and moderator, that the relevant (n, p) and (n, α) reactions follow the $1/\nu$ law, as several studies indicate (42), and that there are 10 ppm nitrogen impurities in the fuel, production rates of ^{14}C have been estimated from reactor design information compiled by IAEA (57). For recent LWRs, the values estimated are, for PWRs, 0.012 Ci per MW(e) y in the fuel and 0.005 Ci per MW(e) y in the moderator and, for BWRs, 0.016 Ci per MW(e) y in the fuel and 0.008 Ci per MW(e) y in the moderator.

75. These results indicate the normalized ^{14}C production to be about 0.02 Ci per MW(e) y. The amount which can be assumed to be released at the reactor, 30 per cent of the total production, is the amount produced in the moderator, which will vary depending on the moderator size. The ^{14}C produced in the fuel will presumably be released, at least partially, at the reprocessing plant.

76. From measurements of decay-tank gas and containment air at three PWRs, Kunz estimated the ^{14}C release in airborne effluents to be approximately 0.006 Ci per MW(e) y (81). In subsequent analyses of off-gas from the main condenser air ejector of a BWR, Kunz estimated the ^{14}C release in airborne effluents to be 0.016 Ci per MW(e) y (82). This result for the R. E. Ginna reactor in the United States corresponds to a somewhat larger moderator size. From detailed measurements at the Oyster Creek BWR in the United States, Blanchard *et al.* (15) estimate the ^{14}C release to be 0.006 Ci per MW(e) y from the main condenser air ejector and 0.002 Ci per MW(e) y from the building ventilation air. Occasional stack monitoring, however, would indicate, if considered representative, a larger airborne release of 0.018 Ci per MW(e) y. Additional measurements at Oyster Creek indicated ^{14}C discharged in liquid effluents to be $2\ 10^{-5}$ Ci per MW(e) y (15).

77. A ^{14}C airborne normalized release rate of 0.013 Ci per MW(e) y was reported for the Yankee Rowe PWR in the United States for 1972 (146). This reactor reported 0.005 Ci per MW(e) y in airborne effluents in 1974 and $3\ 10^{-5}$ Ci per MW(e) y in liquid effluents (163). Some initial measurements of ^{14}C airborne release from reactors operating in the USSR, however, indicate larger airborne releases, namely 0.2, 0.46 and 0.9 Ci per MW(e) y for a PWR, BWR and LWGR, respectively (121). More recent results are somewhat lower. These measurements presumably reflect different reactor design, moderator volume and nitrogen impurity levels.

78. Of the ^{14}C in the BWR off-gas sample measured by Kunz, 95 per cent was in the form of CO_2, 2.5 per cent, CO and 2.5 per cent, hydrocarbons. Due to the reducing atmosphere of the hydrogen cover gas of the PWR, however, over 80 per cent of the ^{14}C activity released from this reactor type was in organic forms CH_4 and C_2H_6 (81, 82).

79. In gas-cooled, graphite moderated reactors the major source of ^{14}C production is the graphite moderator, due to the $^{13}C\,(n,\gamma)\,^{14}C$ reaction and also to the $^{14}N\,(n\ p)\,^{14}C$ reaction based on the incorporated nitrogen impurity. Normalized production rates have been estimated to be about 0.18 Ci per MW(e) y in Magnox reactors and 0.3 Ci per MW(e) y in AGRs (74). Production of ^{14}C in the carbon dioxide coolant, mainly from activation of nitrogen impurities and from the $^{17}O\,(n,\alpha)\,^{14}C$ reaction, is a smaller source estimated to be about $3\ 10^{-3}$ Ci per MW(e) y for Magnox reactors and 10^{-2} Ci per MW(e) y for AGRs (74). Carbon-14 discharges from Magnox reactors and AGRs result from coolant leakage and include ^{14}C released in the coolant from corrosion of the moderator; estimated releases are 0.006 Ci per MW(e) y for Magnox reactors and 0.017 Ci per MW(e) y for AGRs. Assessments of ^{14}C production by HTGRs range from 0.012 Ci per MW(e) y (29) to 0.25 Ci per MW(e) y (90) and originates primarily from the $^{13}C\,(n,\gamma)\,^{14}C$ reaction. Insignificant releases of ^{14}C from these reactors are anticipated. Rather, the ^{14}C would be released at the reprocessing plant (17).

80. The production of ^{14}C in HWRs is due to the same type of nuclear reactions described for LWRs (para. 72). However, due mainly to the large moderator mass, the production rate of ^{14}C in HWRs is expected to be considerably larger than in LWRs. Using the assumptions described in paragraph 74, the production rate in the Atucha reactor in Argentina is estimated to be 0.46 Ci per MW(e) y, with 90 per cent generated in the moderator. The production of ^{14}C in CANDU reactors is estimated to be 0.42 Ci per MW(e) y, 95 per cent being produced in the moderator (120).

81. Much more data, particularly from continuous monitoring, must be forthcoming before a reliable assessment of the ^{14}C generation rate in reactors and its release characteristics can be made.

5. Iodine

82. A number of iodine isotopes are produced in reactors by fission and by decay of other fission products. Release estimates have been published for ^{131}I (8.05 d), ^{132}I (2.3 h), ^{133}I (21 h), ^{134}I (53 m), ^{135}I (6.7 h) and ^{129}I ($1.6\ 10^{7}$ y) (16). Equilibrium amounts of all but ^{129}I are rapidly established in the reactor core, and releases are more a function of the integrity of the coolant system than of the amount of electricity generated. Iodine-131, with a half-life of 8 d, is the main concern from an environmental point of view. Its activity per unit of thermal power at equilibrium in uranium fuel varies from about 25 kCi MW^{-1} at the beginning of the fuel irradiation to about 30 kCi MW^{-1} at the end, as a result of the larger fission yields of plutonium, which contributes increasingly to power production as the burn-up proceeds. Long-lived ^{129}I has not been identified in power reactor

TABLE 12. TRITIUM DISCHARGED IN LIQUID EFFLUENTS FROM REACTORS IN VARIOUS COUNTRIES, 1970-1974

Reactor	*Start-up year*	*Net electrical power (MW(e))*	*Release (Ci)*					*Normalized release (Ci per MW(e) y)*				
			1970	*1971*	*1972*	*1973*	*1974*	*1970*	*1971*	*1972*	*1973*	*1974*
P W R												
France												
SENA, Chooz	1967	270	340	706	1 762	1 849		2.3	3.2	7.2	5.7	
Germany, Fed. Rep. of												
Obrigheim	1968	328		311	243	326	149		1.2	0.89	1.2	0.51
Stade	1972	630			96.7	112	32.2			0.26	0.24	0.05
Biblis	1974	1 147					8.3					0.08
Italy												
Trino, Vercellese	1964	247	135	1 117	1 078			0.95	7.2	4.8		
Japan												
Mihama 1	1970	340	32	140	85	120	47	0.40	0.56	0.67	1.3	1.9
Mihama 2	1972	500			150	190	230			0.63	0.69	0.72
Takahama 1	1974	826					130					0.49
Netherlands												
Borssele	1973	447				1.8	171				0.02	0.53
Sweden												
Ringhals 2	1974	820					6.3					0.27
United States												
Yankee Rowe	1961	175	1 500	1 680	803	694	314	10.3	9.7	10.9	5.8	3.0
Indian Pt. 1	1962	265	410	725	574	138	684	9.4	4.7	4.0	–	5.0
San Onofre	1968	430	4 800	4 570	3 480	4 070	3 810	13.1	11.5	10.8	15.5	10.8
Connecticut Yankee	1968	575	7 400	5 830	5 890	3 900	2 240	17.4	11.6	12.0	14.1	4.5
R. E. Ginna	1970	420	110	154	119	286	195	0.42	0.47	0.43	0.74	0.81
Point Beach, 1,2	1970/72	2 X 497		266	563	556	833		0.68	1.1	0.86	1.1
H. B. Robinson	1971	700		118	405	432	449		0.40	0.81	1.0	0.82
Palisades	1971	700			208	185	8.1			0.96	0.68	0.91
Maine Yankee	1972	790			9.2	154	219			0.17	0.40	0.53
Surry, 1,2	1972/73	2 X 788			5.0	448	245			0.12	0.54	0.36
Turkey Pt. 3,4	1972/73	2 X 693				329	580				0.61	0.64
Indian Pt. 2	1973	873				27.5	47.9				0.69	0.13
Ft. Calhoun	1973	457				15.8	124				0.23	0.45
Prairie Island, 1,2	1973/74	2 X 530				10^{-4}	142				$3.3\ 10^{-5}$	0.89
Oconee 1,2,3	1973/74	3 X 886				70.7	350				0.26	0.55
Zion 1,2	1973	2 X 1 050					2.3					0.0043
Arkansas 1	1974	820					25.6					0.39
Kewaunee	1974	520					92.4					0.51
Three Mile Island	1974	810					130					0.54
		Total	14 727	15 617	15 471	13 905	11 265					
Electrical energy generated (MW(e) y)			1 616	3 124	3 960	6 162	9 069					
Overall normalized release (Ci per MW(e) y)			9.1	5.0	3.9	2.3	1.2					
B W R												
Germany, Fed. Rep. of												
Kahl	1961	15		1.4	1.2	1.7	7.9		0.11		0.30	0.76
Gundremmingen	1966	237		45.6	78.3	148	215		0.20	0.38	0.75	0.98
Lingen	1968	174	31.7		24.0	14.6	9.0	0.28		0.40	0.13	0.17
Würgassen	1972	640			3.2	5.9	3.6			0.05	0.03	0.06
Italy												
Garigliano	1964	152	5.0	5.0	3.0			0.059	0.038	0.062		
Japan												
Tsuruga	1969	331		6.2	5.3	8.2	21.0		0.025	0.020	0.029	0.12

Netherlands												
Dodewaard	1968	52	2.4	1.3	3.2	4.9	4.9	0.056	0.028	0.086	0.12	0.15
Sweden												
Oskarshamn 1	1971	440					10.9					0.071
Ringhals 1	1974	760					2.0					0.23
United States												
Dresden 1	1960	200	5.0	8.7	43.3	18.5	18.8	0.031	0.12	0.34	0.29	0.47
Big Rock Pt.	1962	70	54.0	10.3	10.4	19.7	50.7	1.24	0.27	0.24	0.41	1.31
Humboldt Bay	1963	69	7.0	7.5	13.0	51.3	31.7	0.14	0.19	0.31	1.07	0.73
Lacrosse	1969	53	20.0	91.4	120.0	103.0	115.0	1.31	3.81	4.38	4.56	3.05
Oyster Creek	1969	640	22.0	21.5	61.6	35.9	14.1	0.054	0.047	0.12	0.087	0.033
Nine Mile Pt.	1969	625	20.0	12.4	27.8	46.5	18.7	0.091	0.036	0.075	0.12	0.50
Dresden 2,3	1970/71	2 X 800	31.0	38.5	25.9	25.8	22.6	0.22	0.084	0.028	0.026	0.030
Millstone Pt. 1	1970	652		12.7	20.9	3.7	24.1		0.030	0.058	0.017	0.059
Monticello	1971	545		0.6	0.0001	0	0		0.0036	$2\ 10^{-7}$	0	0
Quad Cities, 1,2	1972	2 X 800			4.7	24.5	34.0			0.011	0.021	0.037
Pilgrim 1	1972	655			4.2	0.4	10.5			0.045	0.00092	0.046
Vermont Yankee	1972	514			0	0.2	0				0.00097	0
Peach Bottom 2,3	1973/74	2 X 1 000					10.0					0.017
Browns Ferry	1973	1 050					2.8					0.0064
Cooper Station	1974	760					1.7					0.0083
		Total	198	263	450	513	629					
Electrical energy generated (MW(e) y)			1 284	2 698	4 072	5 388	5 838					
Overall normalized release (Ci per MW(e) y)			0.15	0.097	0.11	0.095	0.11					
GCR												
Italy												
Latina	1963	153	16.7	13.0	16.9			0.12	0.062	0.12		
United Kingdom												
Berkeley	1962	2 X 138	60.1	43.1	44.2	200	56.7	0.20	0.15	0.17	0.84	0.25
Bradwell	1962	2 X 125	95.3	102.0	251.0	198	117.0	0.45	0.49	1.04	1.08	0.57
Hinkley Pt.	1965	2 X 230	18.6	24.9	38.6	30	39.0	0.11	0.28	0.096	0.11	0.11
Trawsfynydd	1965	2 X 195	67.7	41.9	46.0	116	60.0	0.18	0.11	0.14	0.50	0.16
Dungeness	1965	2 X 205	18.6	35.5	28.9	30.5	20.0	0.051	0.090	0.076	0.083	0.051
Sizewell	1966	2 X 210	20.9	7.0	53.2	208	253	0.049	0.17	0.14	0.63	0.70
Oldbury	1967	2 X 200	17.3	64.4	15.0	13.6	37.4	0.054	0.18	0.048	0.047	0.12
Wylfa	1971	2 X 322	0.55	30.2	82.7	275	134	0.053	0.087	0.25	0.046	0.32
Hunterston	1964	2 X 160	159.0	162.0	37.5	86.7	67.0	0.60	0.63	0.17	0.39	0.28
		Total	475	594	614	1 158	784					
Electrical energy generated (MW(e) y)			2 580	3 004	2 659	2 430	2 887					
Overall normalized release (Ci per MW(e) y)			0.18	0.20	0.23	0.48	0.27					
HWR												
Argentina												
Atucha	1974	320					90.3					0.83
Canada												
Pickering 1,2,3,4	1971/72/73	4 X 508		402	986	5 130	14 400			1.5	3.1	9.0
		Total		402	986	5 130	14 490					
Electrical energy generated (MW(e) y)				344	658	1 630	1 723					
Overall normalized release (Ci per MW(e) y)				1.2	1.5	3.1	8.4					

Sources: References 1, 3, 26, 27, 97, 99, 105, 141, 143, 146, 161, 163, 171.

environs, and its significance, in terms of possible release activities, is much less than that of the other iodine isotopes. However, it is of greater significance in the releases from fuel reprocessing plants and because of its long-term contribution to the collective dose commitment.

83. Iodine measurements at six United States LWRs indicated that on the average 73 per cent of the iodine in reactor off-gas is organic, 22 per cent hypoiodous acid, and 5 per cent elemental (114). All forms were also present in ventilation exhaust air, though the release rate from this source is usually much less than for the reactor off-gas system. The release of iodine nuclides depends strongly on the filter systems used in the installation.

84. Table 13 gives the reported iodine isotope releases in airborne effluents from various reactors. The overall ^{131}I effluent average is (2-5) 10^{-3} Ci per MW(e) y from BWRs and (5-50) 10^{-5} Ci per MW(e) y from PWRs. These values are not necessarily typical for any one reactor. The normalized iodine release amounts for the individual reactors given in table 13 show extremely wide variability. Unusually high results from one or two reactors can easily increase the overall effluent average. The limited data from GCRs and HWRs indicate that the average airborne releases are comparable to that of PWRs.

6. Particulates in airborne effluents

85. Most of the fission and activation nuclides can form aerosols, which may be released with the airborne effluents. Particulates may arise either directly or as decay products of fission noble gases. Aerosols originate particularly from primary coolant leaks, but they can also be generated by working on contaminated components or cleaning contaminated surfaces. High efficiency particulate air (HEPA) filters retain all but the very small fraction of the smallest particles. Released particulate activities are thus very low. Table 14 lists the reported activities and composition of particulates released in airborne effluents from reactors in the United States during 1974 (163). A few additional radionuclides were less frequently identified (163). The average total activities per unit of electrical energy generated are 0.5 mCi per MW(e) y from PWRs and 1.1 mCi per MW(e) y from BWRs. Over 80 per cent of the average PWR release activity is ^{88}Rb, daughter of the noble gas isotope ^{88}Kr. Other small contributors are ^{134}Cs, ^{137}Cs, ^{58}Co, ^{60}Co and ^{54}Mn. For the BWRs, ^{140}Ba + ^{140}La and ^{89}Sr each account for about 30 per cent of the total activity, with ^{131}I, ^{58}Co, ^{60}Co, ^{134}Cs and ^{137}Cs each contributing 10 per cent or less.

7. Radionuclides in liquid effluents

86. Radionuclides other than tritium in liquid effluents include a number of fission and activated corrosion products. The amount and composition of the discharge depend strongly on the type of reactor. The available data on discharges during 1974 are listed in table 15. The average normalized releases, based on operating experience, are approximately 0.008, 0.06 and 0.15 Ci per MW(e) y for PWRs, BWRs and GCRs, respectively. The averages for earlier data (1972 liquid effluents) are similar for BWRs and GCRs but somewhat higher (0.03 Ci per MW(e) y) for PWRs (27, 146). The radionuclides ^{137}Cs and ^{134}Cs account for 30-50 per cent of the total activity discharged by PWRs and GCRs and 70 per cent of that from BWRs; ^{131}I and ^{133}I make up 10-40 per cent and ^{58}Co and ^{60}Co about 15 per cent of the activity in liquid effluents from LWRs. Dissolved noble gases are also present in liquid effluents from LWRs, and in addition several less commonly identified radionuclides have been reported (163).

87. The activity of ^{137}Cs reported as discharged is slightly greater than that of ^{134}Cs. For reactors reporting the release of both caesium isotopes, the ^{134}Cs/^{137}Cs activity ratio is 0.6 for the PWRs and 0.7 for the BWRs. For the GCRs in the United Kingdom the ratio is 0.2. On the basis of an analysis of several years of liquid effluent release data, primarily from reactors in the Federal Republic of Germany, Schiesser estimates that typical normalized releases of ^{137}Cs and ^{134}Cs from large LWRs and 5.0 and 3.1 mCi per MW(e) y, respectively, and of ^{89}Sr and ^{90}Sr, 0.63 and 0.063 mCi per MW(e) y, respectively (125).

88. The iodine isotopes ^{131}I and ^{133}I are released in liquid effluents in amounts comparable to the airborne release from BWRs. Iodine is readily soluble and with little hold-up of liquid releases would be expected to be prevalent in liquid effluents. The iodine isotopes are not reported in liquid discharges from GCRs.

89. The activated corrosion products present in liquid effluents can be expected to vary widely, even among reactors of the same type. In addition to ^{58}Co and ^{60}Co, the isotopes ^{51}Cr and ^{54}Mn are quite prevalent in liquid effluents from LWRs. The reported activities of ^{99}Mo, ^{65}Zn and ^{59}Fe are somewhat less. The isotopes ^{55}Fe and ^{63}Ni are not routinely reported in LWR releases, but Hutchinson *et al.* (56) measured activities comparable to other corrosion products in liquid-waste samples. Blanchard *et al.* (15, 72), however, did not detect ^{63}Ni in liquid effluents from PWRs and BWRs. The Palisades PWR in the United States reported relatively high releases of activated corrosion products, including ^{63}Ni, during 1974, associated with a relatively long maintenance period (163).

90. Most of the activity in liquid effluents from GCRs arise from spent-fuel storage pools. Additional sources are the reactor-vessel cooling system, the coolant-drying installation and various other low-level sources. Liquid streams are collected, treated and released with the condenser coolant water. In GCR liquid effluents, the activation products include ^{35}S, ^{45}Ca and ^{32}P. Activity released in liquid effluents from gas-cooled Magnox reactors (table 15) is somewhat higher than from LWRs. In the United Kingdom all releases are to the sea, except for Trawsfynydd, from which releases are to a fresh-water lake. Liquid wastes are filtered and treated additionally by filtration in non-regenerable ion-exchange beds (60).

TABLE 13. IODINE DISCHARGED IN AIRBORNE EFFLUENTS FROM REACTORS IN VARIOUS COUNTRIES, 1970-1974

	Iodine-131										*Iodine-133*				*Iodine-135*			
	Release (Ci)					*Normalized release (10^{-5} Ci per MW(e) y)*					*Release (Ci)*		*Normalized release (10^{-5} Ci per MW(e) y)*		*Release (Ci)*		*Normalized release (10^{-5} Ci per MW(e) y)*	
Reactor	*1970*	*1971*	*1972*	*1973*	*1974*	*1970*	*1971*	*1972*	*1973*	*1974*	*1972*	*1974*	*1972*	*1974*	*1972*	*1974*	*1972*	*1974*
PWR																		
Belgium																		
BR-3, Mol	0.063	2 10^{-5}	0.001			970		80										
France																		
SENA, Chooz			0.023					9.4										
Germany, Fed. Rep. of																		
Obrigheim	0.045	0.015	0.006	0.005	0.005	15.6	5.8	2.2	1.8	1.7								
Stade			0.047	0.043	0.011			12.7	9.1	1.8								
Biblis					6 10^{-5}					0.06								
Italy																		
Trino, Vercellese	0.0006	0.001	1 10^{-6}			0.42	0.65	0.0004										
Netherlands																		
Borssele				0.0025	0.034				10.4	4.5								
United States																		
Yankee Rowe			0.00023	0.0027	0.00071			0.32	2.25	0.68		0.00029		0.28		0.00039		0.37
Indian Pt. 1			0.058	0.012	0.09			40.8		65.7		0.0045		3.25				
San Onofre			4.4 10^{-5}	0.42	0.00019			0.013	160	0.053		1.2 10^{-5}		0.0034		3.1 10^{-5}		0.0088
Connecticut Yankee			0.01	0.0013	4.6 10^{-8}			2.04	0.47	9.1 10^{-6}	6.3 10^{-5}		0.013					
R. E. Ginna			0.034	0.00053	0.00028			12.1	0.14	0.12	0.0015	0.00017	0.54	0.069				
Point Beach 1,2			0.0097	0.011	0.088			1.82	1.69	11.5		0.038		5.02				
H. B. Robinson				0.30	0.046				68.2	8.47		0.0034		0.63		0.00095		0.17
Palisades			0.0087	0.29	0.0099			4.0	104	112		0.010		112		0.0019		21.8
Maine Yankee			1.7 10^{-6}	0.0016	0.054			0.0032	0.41	13.2	0.0042	0.0019	7.9	0.46	0.0011		2.1	
Surry 1,2			1.2 10^{-5}	0.042	0.10			0.029	5.04	15.0	4.5 10^{-5}	0.020	0.11	2.90		0.0042		0.63
Turkey Pt. 2,3				0.055	3.41				10.3	378		0.034		3.74		0.0052		0.57
Indian Pt. 2				0.00029	0.37				0.73	98.0		0.056		14.9				
Ft. Calhoun				4.3 10^{-5}	0.00042				0.063	0.15		5.0 10^{-5}		0.018		4.2 10^{-5}		0.015
Prairie Island				1.3 10^{-5}	0.00045				0.58	0.28		0.00014		0.086		1.6 10^{-5}		0.010
Oconee 1,2				0.012	0.032				4.47	5.01		0.00037		0.059		8.1 10^{-6}		0.0013
Zion 1,2				6.7 10^{-7}	0.014				0.00083	2.65		0.0011		0.21				
Arkansas 1					0.053					81.4								
Kewaunee					0.024					13.1		0.0027		1.46		0.0023		1.25
Three Mile Island					0.028					1.18		0.00013		0.052		2.0 10^{-7}		8.2 10^{-5}
Total	0.11	0.02	0.20	1.20	4.37						0.058	0.17			0.011	0.015		
Electrical energy generated (MW(e) y)	438	412	3 133	5 471	8 436						863	6 544			53	4 079		
Overall normalized release (10^{-5} Ci per MW(e) y)	25	3.9	6.3	22	52						0.67	2.6			2.1	0.37		

TABLE 13 *(continued)*

	Iodine-131										*Iodine-133*				*Iodine-135*			
	Release (Ci)					*Normalized release (10^{-3} Ci per MW(e) y)*					*Release (Ci)*		*Normalized release (10^{-3} Ci per MW(e) y)*		*Release (Ci)*		*Normalized release (10^{-3} Ci per MW(e) y)*	
Reactor	*1970*	*1971*	*1972*	*1973*	*1974*	*1970*	*1971*	*1972*	*1973*	*1974*	*1972*	*1974*	*1972*	*1974*	*1972*	*1974*	*1972*	*1974*
BWR																		
Germany, Fed. Rep. of																		
Kahl	0.005	0.0025		0.002	0.003	0.38	0.19		0.32	0.29								
Gundremmingen	0.20	0.34	0.20	1.96	0.12	0.94	1.5	0.94	9.9	0.53								
Lingen	0.26	0.38	0.15	0.016	0.002	2.3	3.3	2.5	0.15	0.038								
Würgassen			$< 10^{-4}$	$< 10^{-4}$	< 0.0007			< 0.002	< 0.0004	< 0.01								
Italy																		
Garigliano	0.06	0.13	0.06			0.71	0.98	1.2										
Japan																		
Tsuruga	0.38	1.06	0.24	0.20	0.28	1.4	4.2	0.92	0.70	1.6								
Fukushima 1,2		0.06	0.25	0.79	0.34		0.19	0.84	3.5	0.69								
Netherlands																		
Dodewaard	0.0005	0.009	0.0015	0.001	0.0095	0.012	0.020	0.040	0.024	0.03								
United States																		
Dresden 1				0.04	0.41				1.1	10.3		4.41		110		8.66		220
Big Rock Pt.			0.12	4.6	0.09			2.8	96.0	2.3	0.017	0.26	0.39	6.8		0.00063		0.016
Humboldt Bay			0.40	0.17	0.50			9.7	3.5	11.6	1.7	1.19	41.1	27.4		0.012		0.28
Lacrosse			0.69	0.18	0.033			25.3	8.0	0.87	0.47	0.019	17.2	0.49	0.012		0.44	
Oyster Creek			6.26	6.7	2.31			12.6	16.0	7.8	7.04	10.5	14.1	24.9	5.82	9.54	11.7	22.6
Nine Mile Pt.			0.89	1.96	0.72			2.4	4.9	1.9	0.55	1.66	1.5	4.4	0.28	0.17	0.76	0.45
Dresden 2,3				0.49	3.98			0.99		5.3		14.7		19.5		19.3		25.6
Millstone 1			1.23	0.15	3.18			3.4	0.70	7.7								
Monticello			0.58	1.2	5.63			1.4	3.2	17.2	1.16	11.0	2.9	33.2	2.26	12.0	5.6	36.2
Quad Cities 1,2			0.73	5.5	8.82			1.6	4.7	9.5	0.60	16.0	1.4	17.3	0.070	11.8	0.16	12.8
Pilgrim 1			0.03	0.46	1.44			0.3	0.97	6.3	0.92	1.12	9.9	4.9		1.87		8.2
Vermont Yankee			0.17	0.06	0.35			3.5	0.29	1.2	0.16	0.12	3.3	0.42				
Peach Bottom 2,3					0.0066					0.11								
Browns Ferry 1					0.38					0.086		0.045		0.10		0.41		0.92
Cooper Station					0.13					0.062		0.086		0.42		3.44		16.7
Total	0.91	1.98	12.0	24.5	29.7						12.6	61.1			8.4	67.2		
Electrical energy generated (MW(e) y)	747	1 094	3 312	5 509	6 134						1 972	4 129			1 746	3 806		
Overall normalized release (mCi per MW(e) y)	1.2	1.8	3.6	4.4	4.8						6.4	15.0			4.8	18.0		

	Iodine-131										*Iodine-133*				*Iodine-135*			
	Release (Ci)					*Normalized release (10^{-5} Ci per MW(e) y)*					*Release (Ci)*		*Normalized release (10^{-5} Ci per MW(e) y)*		*Release (Ci)*		*Normalized release (10^{-5} Ci per MW(e) y)*	
Reactor	*1970*	*1971*	*1972*	*1973*	*1974*	*1970*	*1971*	*1972*	*1973*	*1974*	*1972*	*1974*	*1972*	*1974*	*1972*	*1974*	*1972*	*1974*
G C R																		
France																		
Chinon			0.027					5.4										
St. Laurent			0.065					9.9										
Bugey			0.0001					0.08										
Total			0.09															
Electrical energy generated (MW(e) y)			1 285															
Overall normalized release (10^{-5} Ci per MW(e) y)			7.2															
H W R																		
Argentina																		
Atucha					0.0098					9.1								
Canada																		
Pickering			0.13	0.0055	0.0040			20.0	0.34	0.25								
Total			0.13	0.0055	0.0138													
Electrical energy generated (MW(e) y)			658	1 630	1 723													
Overall normalized release (10^{-5} Ci per MW(e) y)			20.0	0.34	0.80													

Sources: References 1, 26, 27, 43, 44, 99, 146, 161, 163, 171.

TABLE 14. PARTICULATES IN AIRBORNE EFFLUENTS FROM UNITED STATES REACTORS

Reactor	*Electrical energy generated in 1974 (MW(e) y)*	*Amount released (mCi) during 1974*											
		^{88}Rb	^{137}Cs	^{54}Mn	^{58}Co	^{60}Co	^{134}Cs	^{59}Fe	^{89}Sr	^{138}Cs	^{51}Cr	^{140}Ba	^{90}Sr
P W R													
Yankee Rowe	103.9		0.016	0.41	0.95	2.18	0.0081		0.74		0.5	0.52	0.11
Indian Pt. 1	137.0	542	7.26	0.84	0.85	7.05	2.74	0.047	0.052				0.023
Connecticut Yankee	502.3	0.16											
R. E. Ginna	239.7		0.020	$6.3 \; 10^{-4}$	0.0023	0.012	0.0049	$2.5 \; 10^{-4}$			$7.8 \; 10^{-4}$		
Pt. Beach 1,2	764.8	0.92	0.23		28.0	37.2				0.14			
H. B. Robinson	547.9			0.024	0.70	0.36	0.066					0.091	
Palisades	8.9		0.12	0.60	2.0	0.58	0.042						
Maine Yankee	411.0		0.016		0.41	0.19	$9.3 \; 10^{-4}$				0.0046		
Surry 1,2	673.5		1.71	37.4	1.16	1.16	0.0001						
Turkey Pt. 3,4	901.8		88	2.91	17.7	15.8	52	0.018	0.092			0.11	0.012
Indian Pt. 2	376.7	170	0.53	7.67	41.6	7.06	0.43	3.08	1.05				0.27
Ft. Calhoun	274.0				0.023				$3.7 \; 10^{-4}$			0.011	$3.4 \; 10^{-4}$
Oconee 1,2,3	632.4		0.032		0.070	0.010	$5.5 \; 10^{-4}$		0.012	0.40	0.0039	0.042	$1.2 \; 10^{-7}$
Zion 1,2	536.5		0.040		0.27				0.0005				0.034
Arkansas 1	65.1								0.0025				
Kawaunee	182.6	136			0.37					0.14			
Three Mile Island	239.7				0.0023				$3.4 \; 10^{-4}$				$7.1 \; 10^{-4}$
Total		849	98.0	49.9	94.1	71.6	55.3	3.1	1.9	0.68	0.51	0.78	0.46
Corresponding electrical energy (MW(e) y)		1 963	4 786	2 989	6 030	4 798	4 033	1 655	3 267	1 580	1 387	2 460	3 202
Normalized release (μCi per MW(e) y)		430	21.0	17.0	16.0	15.0	14.0	1.9	0.60	0.43	0.37	0.32	0.14
		^{140}Ba	^{89}Sr	^{131}I	^{60}Co	^{134}Cs	^{51}Cr	^{137}Cs	^{58}Co	^{54}Mn	^{90}Sr	^{59}Fe	^{65}Zn
B W R													
Dresden 1	40.0	74.4	78.1	32.9	45.0			3.6	5.5		1.0		
Big Rock Pt.	38.8				5.23	1.59	7.69	7.79		1.29			0.040
Humboldt Bay	43.4	126	202		0.35	0.27		4.47		0.47	0.17		0.14
Lacrosse	37.7	1.2	4.7			1.2		1.2			1.5		
Oyster Creek	422.4	58.1	47.4	33.4	8.4	1.7	1.5	2.6	1.6	3.3	1.2	0.20	
Nine Mile Pt.	376.7		3.65		7.54	3.66	4.58	7.41	0.16	1.7	4.61	0.18	0.056
Dresden 2,3	753.4	886	991	334	267	15.6	49.2	33.2	47.4	9.3	11.0		
Millstone Pt.	411.0		3.35	29.6	12.9	2.76	2.68	3.67	0.79	6.13	0.41	0.63	0.046
Quad Cities 1,2	924.7	6.9	5.83	34.0	3.5	3.44	6.1	4.9	0.65	0.71	0.095	0.0056	0.21
Pilgrim 1	228.3	4,7	0.89		1.38	0.15	0.014	0.44	0.31	0.99	0.0055	0.098	0.052
Vermont Yankee	285.4	1.63			0.031	1.1		0.78		0.0077			0.089
Browns Ferry	593.6	35.9	0.16			38.4		9.15			0.030		
Cooper Station	205.5	195	0.023			28.8					0.012		
Total		1 390	1 337	464	351	98.7	71.8	79.2	56.4	23.9	20.0	1.1	0.63
Corresponding electrical energy (MW(e) y)		3 534	4 037	2 552	3 524	4 321	3 155	4 155	3 157	3 484	4 037	2 363	2 308
Normalized release (μCi per MW(e) y)		390	330	180	100.0	23.0	23.0	19.0	18.0	6.9	5.0	0.47	0.27

Source: Reference 163.

TABLE 15. RADIONUCLIDES OTHER THAN TRITIUM DISCHARGED IN LIQUID EFFLUENTS FROM VARIOUS REACTORS DURING 1974

Reactor	*Electrical energy generated (MW(e) y)*	*Amount discharged (Ci)*																		
		^{131}I	^{137}Cs	^{133}I	^{134}Cs	^{58}Co	^{60}Co	^{51}Cr	^{54}Mn	^{124}Sb	^{144}Ce	^{103}Ru	^{110m}Ag	^{89}Sr	^{132}I	^{59}Fe	^{95}Nb	^{95}Zr	^{99}Mo	^{90}Sr
P W R																				
Germany, Fed. Rep. of																				
Obrigheim	293.5	0.083	0.61		0.40	0.64	0.27	0.019	0.073	0.018				0.0056						0.001
Stade	608.2	0.0007	0.041		0.013	0.013	0.015	0.0046	0.0043	0.020	0.0025	0.0010	0.027	0.0011			0.0029	0.0011		0.0004
Biblis	100.8	0.0011				0.040	0.0032		0.0034	0.27				0.0001						9 10^{-6}
United States																				
Yankee Rowe	103.9	0.0006	0.0032	5 10^{-5}	0.0020	0.0001	0.0003	5 10^{-5}	0.0003				4 10^{-6}	0.0003						7 10^{-5}
Indian Pt. 1	137.0	0.40	0.45		0.20	0.12	0.16							0.0062		0.0077				0.0008
San Onofre	353.9	0.047				0.35	0.070	0.19	0.011	0.013			0.017	0.0020	0.032	0.018				0.0044
Conn. Yankee	502.3	9 10^{-5}	0.27		0.0030	0.42	0.68	0.51	0.0045		0.28	0.048								
Ginna	239.7	0.0001	0.039		0.013	0.014	0.032		0.0034		0.015	0.0040	0.0002					0.0064		
Pt. Beach 1,2	764.8	0.018	0.098	0.029	0.038	0.0013	0.0002		2 10^{-5}		8 10^{-6}	0.0005		0.0009	0.0012		0.0001	8 10^{-5}		0.0002
H. B. Robinson	547.9	0.82	0.15		0.14	0.22	0.16	0.0081	0.022	0.0002			0.0003	0.0032		2 10^{-5}			8 10^{-5}	0.0011
Palisades	8.9	0.0056	1.48	0.0005	0.70	0.70	0.43	0.014	0.29							0.0012	0.0076	0.0021		
Maine Yankee	411.0		0.99	0.14	1.49	0.27			0.0025									0.0016		1 10^{-5}
Surry 1,2	673.5	11.0	6.83	4.67	3.73	1.93	1.00		0.36					0.020						0.0015
Turkey Pt. 3,4	901.8	1.07	0.068	0.099	0.036	0.16	0.029	0.021	0.0082	0.019	0.0077			0.0094	0.0007	0.0026		0.0028	0.0051	0.0002
Indian Pt. 2	376.7	2.18	0.24		0.056	1.34	0.14		0.062					0.0019		0.012				0.0003
Ft. Calhoun	274.0			0.017		0.14	0.0029		0.011					4 10^{-5}						7 10^{-5}
Prairie Island	160.0					0.0003	4 10^{-5}													
Oconee 1,2,3	632.4	0.11	0.060	0.092	0.040	0.47	0.028	0.017	0.0077	0.0002	0.0005		1 10^{-6}	0.016	0.0030	3 10^{-5}	0.0002	0.0001	0.0029	0.0020
Zion 1,2	536.5					0.0008														
Arkansas 1	65.1	0.019	0.0002	0.0003		0.033	0.0006	0.0091	0.0021		0.0015	6 10^{-7}	0.0013	0.017	0.0001	0.0018	0.0003	0.0002	2 10^{-5}	0.0025
Kewaunee	182.6	0.021	0.0072	0.0027	0.0011	0.29	0.017	0.0045	0.0089	0.0090					6 10^{-5}	0.0001	0.0024	0.0017	0.0007	
Three Mile Island	239.7	0.0092	0.0035		3 10^{-5}	0.018	0.0001	0.0053	0.0002					2 10^{-5}				0.0003	4 10^{-6}	2 10^{-6}
Total		15.78	11.34	5.05	6.85	7.17	3.03	0.803	0.874	0.354	0.304	0.0531	0.0458	0.0835	0.0370	0.0437	0.0135	0.0165	0.0088	0.0144
Corresponding electrical energy (MW(e) y)		6 733	6 689	4 018	6 624	8 114	7 167	4 440	7 281	3 621	3 714	2 180	2 551	6 073	2 836	3 206	2 262	4 054	2 570	6 484
Normalized release (mCi per MW(e) y)		2.3	1.7	1.3	1.0	0.88	0.42	0.18	0.12	0.098	0.082	0.024	0.018	0.014	0.013	0.013	0.0060	0.0041	0.0034	0.0022

		^{137}Cs	^{134}Cs	^{60}Co	^{131}I	^{54}Mn	^{58}Co	^{89}Sr	^{51}Cr	^{144}Ce	^{65}Zn	^{24}Na	^{99}Mo	^{133}I	^{90}Sr	^{59}Fe	^{95}Zr	^{141}Ce	^{110m}Ag
B W R																			
Germany, Fed. Rep. of																			
Kahl	10.4	0.0017	0.0002	0.0028	0.0002			2 10^{-5}							0.0010				
Gundremmingen	219.1	0.12	0.059	0.040	0.11	0.0098	0.057	0.53		0.0007	6 10^{-5}				0.052	0.0015	0.0006	0.0030	0.011
Lingen	54.9	0.0044	0.00012	0.0057				0.0021							3 10^{-5}	1 10^{-5}			
Würgassen	55.7	0.0005	0.0005	0.36		0.014	0.16	0.0028	0.10		0.33				2 10^{-5}	0.0004	0.0041	0.0004	0.0009
United States																			
Dresden 1	40.0	1.37	0.67	0.72	0.034	0.036	0.0060	0.0098						0.023	0.0019	0.0022			
Big Rock Pt.	38.8	0.18	0.075		0.0052	0.049					0.021			0.0002		0.010			
Humboldt Bay	43.3	2.10	1.37	0.28	0.034	0.23	0.0031	0.074		0.12	0.14			0.0015	0.018				
Lacrosse	37.7	2.72	1.51	0.53		0.079	6.81	0.013	0.061		0.096				0.0012				
Oyster Creek	422.4	0.015	0.026	0.074	0.014	0.044	0.0092	0.012	0.11	0.0031	0.0020		0.11	0.016	0.0062	0.012	0.0077	0.0041	
Nine Mile Pt.	376.7	9.78	4.64	5.16	1.01	2.46	0.015	0.85	0.27			0.17	0.36	0.48	0.12	0.010	0.029	0.0080	
Dresden 2,3	753.4	6.79	2.77	12.3	0.55	0.72	0.23	0.62	0.059	0.65			0.0016	0.0032	0.028		0.0023		
Millstone Pt.	411.0	95.2	72.1	9.67	13.4	6.18	0.50	1.62			0.17				0.21	0.13			
Monticello	331.1	no liquid releases																	
Quad Cities 1,2	924.7	6.00	2.95	3.01	2.09	0.38	0.43	0.24	1.10		0.12			0.0087	0.045	0.0002	0.0008		8 10^{-5}

TABLE 15 *(continued)*

Reactor	*Electrical energy generated (MW(e) y)*	*Amount discharged (Ci)*																	
		^{137}Cs	^{134}Cs	^{60}Co	^{131}I	^{54}Mn	^{58}Co	^{89}Sr	^{51}Cr	^{144}Ce	^{65}Zn	^{24}Na	^{99}Mo	^{133}I	^{90}Sr	^{59}Fe	^{95}Zr	^{141}Ce	^{110m}Ag
United States *(continued)*																			
Pilgrim I	228.3	1.58	0.50	0.96	0.17	0.28	0.30	0.016	0.12	0.0008	0.056		0.010	0.0011	0.0027	0.023	0.0010	7 10^{-5}	0.0002
Vermont Yankee	285.4	no liquid releases																	
Browns Ferry 1	593.6	0.0011	0.0081	0.18	0.10	0.014	0.014	0.0052	0.091		0.30	0.017			0.0051	0.016			
Peach Bottom 2,3	441.8			0.0075		0.0052	0.045		0.069		0.0030	0.21	0.0018			0.0006	0.0034		
Cooper Station	205.5	0.0019	0.22	0.077	0.0028	0.0024	0.57	0.0098	0.72		0.020		0.010		0.0009	0.0087	0.0020	0.0038	
	Total	125.87	86.90	33.21	17.44	10.50	9.13	4.00	2.70	0.775	0.994	0.399	0.497	0.528	0.490	0.212	0.0510	0.0194	0.0121
Corresponding electrical energy (MW(e) y)		5 032	5 032	5 435	4 884	5 409	5 370	4 993	4 656	2 283	4 239	2 029	3 045	3 444	4 993	4 629	4 244	2 124	2 044
Normalized release (mCi per MW(e) y)		25.0	17.3	6.1	3.6	1.9	1.7	0.80	0.58	0.34	0.23	0.20	0.16	0.15	0.098	0.046	0.012	0.0091	0.0059

		^{137}Cs	^{35}S	^{134}Cs	^{90}Sr	^{90}Y	^{125}Sb	^{147}Pm	^{55}Fe	^{106}Ru	^{106}Rh	^{125m}Te	^{45}Ca	^{144}Ce
G C R														
United Kingdom														
Berkeley	224.9	11.3	3.26	2.19	1.99	1.99	0.092	0.25	0.12	0.092	0.092	0.023	0.85	0.069
Bradwell	204.3	27.5	7.11	3.33	16.6	16.6	0.36	4.50	4.86	1.71	1.71	0.090	0.63	1.08
Hinkley Pt.	350.5	61.8	4.13	5.63	6.25	6.25	16.0	4.75	2.13	4.00	4.00	4.13	0.25	1.25
Trawsfynydd	369.9	2.20	11.1	0.32	0.17	0.17	3.14	0.095	0.019	0.30	0.30	0.76	0.019	0.019
Dungeness	388.1	43.0	6.07	13.0	1.45	1.45	0.069	0.14	1.86	< 0.035	< 0.035	< 0.035	0.62	0.069
Sizewell	360.7	6.49	4.69	1.10	0.70	0.70	0.048	0.095	0.59	0.048	0.048	0.016	0.52	0.016
Oldbury	320.8	12.6	6.68	2.28	4.24	4.24	0.16	0.098	0.20	0.033	0.033	0.033	0.52	0.033
Wylfa	424.7	0.036	0.26	0.008	0.024	0.024	0.001	0.008	0.056	0.004	0.004	0.001	0.001	0.004
Hunterston	243.2	32.5	8.58	11.4	1.94	1.94							0.41	
	Total	197.3	51.84	39.24	33.32	33.32	19.87	9.94	9.82	6.19	6.19	5.05	3.83	2.54
Corresponding electrical energy (MW(e) y)		2 887	2 887	2 887	2 887	2 887	2 644	2 644	2 644	2 664	2 664	2 664	2 887	2 644
Normalized release (mCi per MW(e) y)		68.3	18.0	13.6	11.5	11.5	7.5	3.8	3.7	2.3	2.3	1.9	1.3	0.96

		^{144}Pr	^{32}P	^{60}Co	^{89}Sr	^{51}Cr	^{124}Sb	^{91}Y	^{154}Eu	^{95}Nb	^{14}C	^{65}Zn	^{54}Mn	^{59}Fe
Berkeley	224.9	0.069	0.069	0.023	0.28	0.16	0.069	0.046	< 0.012	0.023	0.046	< 0.012	< 0.012	< 0.012
Bradwell	204.3	1.08	0.72	1.53	< 0.045	< 0.045	0.18	< 0.045	0.27	< 0.045	< 0.045	0.18	0.09	< 0.045
Hinkley Pt.	350.5	1.25	0.25	0.13	0.38	1.00	0.25	0.88	0.13	0.13	0.13	< 0.063	< 0.063	< 0.063
Trawsfynydd	369.9	0.019	0.25	0.038	< 0.0095	< 0.0095	0.057	< 0.0095	< 0.0095	0.019	< 0.0095	0.019	< 0.0095	0.019
Dungeness	388.1	0.069	0.21	0.069	0.21	0.35	0.35	< 0.035	<0.035	0.069	< 0.035	< 0.035	< 0.035	< 0.035
Sizewell	360.7	0.016	0.43	0.048	0.080	0.11	0.064	0.032	< 0.008	0.016	< 0.0080	0.016	0.016	0.016
Oldbury	320.8	0.033	0.13	0.033	0.85	0.13	0.13	0.033	< 0.016	0.033	0.065	< 0.016	< 0.016	0.033
Wylfa	424.7	0.004	< 0.0003	0.008	0.001	0.016	0.004	0.001	< 0.0003	0.004	0.001	0.004	0.004	0.005
Hunterston	243.2		0.18											
	Total	2.54	2.23	1.87	1.79	1.76	1.10	0.987	0.395	0.289	0.237	0.219	0.110	0.725
Corresponding electrical energy (MW(e) y)		2 644	2 887	2 644	2 644	2 644	2 644	2 644	2 644	2 644	2 644	2 644	2 644	2 644
Normalized release (mCi per MW(e) y)		0.96	0.77	0.71	0.68	0.67	0.42	0.37	0.15	0.11	0.090	0.083	0.042	0.027

Sources: References 3, 106, 163.

B. LOCAL AND REGIONAL COLLECTIVE DOSE COMMITMENT

91. Routine environmental monitoring is conducted to a greater or lesser extent around all reactor installations. More detailed, investigative studies to characterize reactor environmental contamination caused by releases have been conducted around specific types of reactors (15, 70, 71, 72). Except in the immediate vicinity of the reactor, the levels of radioactive contamination are not usually readily detectable. Dose assessments must therefore rely primarily on calculations of the environmental dispersion of the released radioactivity. Generalized calculations can be expected to indicate roughly the relative significance of the release amounts and of the exposure pathways. More specific dose assessments at a given site may require a different selection of model parameters to fit the local conditions.

1. Fission noble gases

92. The noble gases released from reactors usually make the largest contribution to the dose to the local population. Considerable effort has been devoted to the development and application of calculational methods to estimate the airborne dispersion (23, 96), and in a few cases careful attempts to measure the actual exposure from the noble gas plume have been successful (9, 37, 98).

93. Bryant has calculated the gonad dose per unit activity discharged that would be received by the population at distances from 1 to 1000 km for each of noble gas nuclides (23). The calculations assume a 30 m effective stack height and average weather conditions. Assuming a population density of 100 km^{-2} in the region from 1 to 100 km and 25 km^{-2} at greater distances, the corresponding collective doses per unit activity discharge have been calculated (table 16). When these results are applied to actual or generalized released amounts, the total dose and the relative contributions of the various nuclides can be obtained. The specific noble gas release composition can vary substantially, depending on how the waste gas is processed.

94. Using the average release composition based on operating experience (table 9), the examples of table 17 have been calculated. In the computation for the BWR given in table 17, the gonad doses at 1 and 10 km are due primarily to ^{88}Kr, ^{87}Kr, ^{138}Xe and ^{135}Xe. The ^{88}Kr daughter, ^{88}Rb, although largely prevented from being released by the particle filters, builds up as the parent decays and is an important dose contributor at 10 km. The same holds for the ^{138}Xe daughter, ^{138}Cs. At 100 km, ^{135}Xe (9.2 h) becomes the dominant contributor. The fractional contribution from the relatively long-lived ^{133}Xe (5.3 d) increases with distance. At 1000 km, only ^{135}Xe and ^{133}Xe make significant dose contributions. Estimates of the corresponding collective doses per unit of noble gas activity released are 4.5 10^{-4} man rad Ci^{-1} within 100 km, and an additional 0.4 10^{-4} man rad Ci^{-1} in the region from 100-1000 km. For the average BWR total normalized release based on operating experience, 1100 Ci per MW(e) y (table 8), the local collective dose would be 0.5 man rad per MW(e) y (to 100 km), and the additional regional contribution would be 0.05 man rad per MW(e) y (100-1500 km), including the contribution from ^{133}Xe beyond 1000 km.

TABLE 16. GONADAL DOSE FACTORS FOR NOBLE GASES RELEASED IN AIRBORNE EFFLUENTS[a]

		Dose per unit release (10^{-9} rad Ci^{-1})				*Collective dose per unit release (10^{-3} man rad Ci^{-1})*[b]		
Radionuclide	*Half-life*	*1 km*	*10 km*	*100 km*	*1000 km*	*1-10 km*	*10-100 km*	*100-1000 km*
$^{83\text{m}}$Kr	1.86 h	3.2	0.16			0.014	0.015	
$^{85\text{m}}$Kr	4.4 h	18.4	0.92	0.010		0.069	0.18	0.0078
^{85}Kr	10.76 y	0.18	0.012	0.00076	0.000048	0.00080	0.0050	0.0080
^{87}Kr	76 m	66.6	2.2	0.0016		0.20	0.20	
^{88}Kr	2.8 h	101.0	5.4	0.025		0.40	0.78	0.012
^{88}Rb[c]	17.8 m	8.2	1.7	0.0079		0.095	0.29	
^{89}Kr	3.2 m	76.1	0.0032					
^{89}Rb[d]	15.4 m	23.8	0.27					
$^{131\text{m}}$Xe	11.8 d	13.3	0.76	0.044	0.0023	0.055	0.31	0.41
$^{133\text{m}}$Xe	2.26 d	22.5	1.3	0.067	0.0011	0.092	0.47	0.46
^{133}Xe[e]	5.27 d	0.0014	0.00079	0.00043	0.00012	0.000028	0.0016	0.014
^{133}Xe	5.27 d	2.3	0.14	0.0065	0.00026	0.0094	0.050	0.054[f]
$^{135\text{m}}$Xe	15.6 m	38.1	0.24			0.052	0.0051	
^{135}Xe	9.2 h	25.4	1.3	0.032	0.00029	0.097	0.34	0.11
^{137}Xe	3.9 m	4.8	0.00041					
^{138}Xe	17 m	72.9	0.70			0.16	0.018	
^{138}Cs[g]	32.2 m	12.0	1.3	0.000070		0.098	0.066	
^{41}Ar	1.83 h	85.6	3.2	0.0057		0.26	0.22	0.0023

Source: Reference 23.

[a] Effective stack height of 30 m; average weather conditions.

[b] Assumed 100 man km^{-2} in the region 1-100 km and 25 man km^{-2} at distances > 100 km.

[c] Per unit activity of ^{88}Kr released.

[d] Per unit activity of ^{89}Kr released.

[e] Per unit activity of $^{133\text{m}}$Xe released.

[f] The extrapolated collective dose in the region 1000-1500 km, per unit activity of ^{133}Xe released, is 0.017 10^{-3} man rad Ci^{-1}.

[g] Per unit activity of ^{138}Xe released.

TABLE 17. GONADAL DOSE FROM NOBLE GASES RELEASED IN AIRBORNE EFFLUENTS FROM REACTORS[a]

	Dose per unit release (10^{-9} rad Ci^{-1})				*Collective dose per unit release (10^{-3} man rad Ci^{-1})*		
Type of reactor	*1 km*	*10 km*	*100 km*	*1 000 km*	*1-10 km*	*10-100 km*	*100-1 000 km*
B W R	44	2.0	0.015	0.00012	0.16	0.29	0.041
P W R	5.7	0.26	0.0079	0.00027	0.020	0.070	0.059
G C R	86	3.2	0.0057		0.26	0.22	0.0023

		Relative contribution to dose (%)				*Relative contribution to collective dose (%)*		
Radio-nuclide	*Fraction of total release (%)*	*1 km*	*10 km*	*100 km*	*1 000 km*	*1-10 km*	*10-100 km*	*100-1 000 km*
B W R								
^{135}Xe	26	15	17	55	63	16	30	70
^{88}Kr	14	32	39	23		35	37	4.1
^{88}Rb[b]		2.6	12	7.4		8.3	14	
^{133}Xe[c]	14	0.7	1	6.1	30	0.8	2.4	18
^{138}Xe	13	21	4.7			13	0.8	
^{138}Cs[b]		3.5	8.7	0.06		7.9	2.9	
^{87}Kr	12	18	14	1.3		15	8.2	
^{83m}Kr	7	0.5	0.6			0.6	0.4	
^{85m}Kr	5	2.1	2.4	3.3		2.2	3.1	1.0
^{135m}Xe	4	3.5	0.5			1.3	0.07	
^{85}Kr	2	0.008	0.01	0.1	0.8	0.01	0.03	0.4
^{137}Xe	0.9	0.1	0.0002					
^{133m}Xe	0.6	0.3	0.4	2.7	5.5	0.3	1.0	6.7
^{133}Xe[b]			0.0003	0.02	0.6		0.003	0.2
P W R								
^{133}Xe[c]	90	37	48	74	86	42	64	82
^{135}Xe	3	13	15	12	3.2	15	14	5.6
^{138}Xe	1	13	2.7			8.0	0.3	
^{85}Kr	1	0.03	0.05	0.1	0.2	0.04	0.07	0.1
^{131m}Xe	1	2.4	2.9	5.6	8.5	2.8	4.4	6.9
^{88}Kr	0.9	16	19	2.8		18	10	0.2
^{135m}Xe	0.8	5.4	0.7			2.1	0.06	
^{85m}Kr	0.6	2.0	2.1	0.8		2.1	1.5	0.08
^{133m}Xe	0.5	2.0	2.5	4.2	2.0	2.3	3.3	3.9
^{133}Xe[b]			0.002	0.03	0.2		0.01	0.1
^{87}Kr	0.4	4.7	3.4	0.08		4.0	1.1	
^{41}Ar	0.3	4.5	3.7	0.2		3.9	0.9	0.01
G C R								
^{41}Ar	100	100	100	100	100	100	100	100

[a] The dose factors of table 16 were used; therefore, the results apply to a 30-m effective stack height and average weather conditions.

[b] Formed after release and decay of parent radionuclide.

[c] The extrapolated collective dose in the region 1 000-1 500 km due to ^{133}Xe is 0.0024 10^{-3} and 0.015 10^{-3} man rad Ci^{-1} for the BWR and PWR, respectively.

95. Using reported noble gas release data and actual average local meteorology and population factors, Martin has computed the collective dose to the population within 80 km of several BWRs in the United States (96). The results for 1971 ranged from 0.04 to 4.0 man rad per MW(e) y for eight reactor sites. The average noble gas release was 1718 Ci per MW(e) y. The collective dose was 900 man rad from 1.85 GW(e) y of electrical energy generated, corresponding to 0.5 man rad per MW(e) y. Approximately similar results were obtained for 11 BWR sites in 1972 (94). Preliminary results for 1973 indicated a probable slight reduction in the collective dose due to the longer hold-up times of the newer BWRs (95). The results of Martin, which are based on actual population and meteorology data, are in good agreement with the estimates of the previous paragraph.

96. Attempts to evaluate the absorbed dose rates in air from released noble gas by actual measurements within a few kilometres of BWRs have met with success (9, 98). Sensitive ionization chambers clearly register the overhead passage of the plume; however, some care is needed to distinguish similar abrupt changes in the background exposure rate due to changes in the radon-daughter contribution during and shortly after a rainfall.

97. Gogolak has used the results of a monitoring network of continuously operating ionization chambers to evaluate model dispersion calculations (37). Seven ionization chambers were placed at distances of 1.3-8 km from a BWR located at a sea-coast site. In the most prominent wind direction, the absorbed dose in air per unit of activity released, measured over a 200-d period, averaged $4.2\ 10^{-9}$ rad Ci^{-1} at 2.6 km and $1.5\ 10^{-9}$ rad Ci^{-1} at 8 km. The computed absorbed doses in air, with careful correlation of stack-effluent release monitoring and local meteorology, agreed very well with the measurements. For the actual population around the reactor, Gogolak determined the whole-body collective dose per unit of electrical energy generated to be 0.02 man rad per MW(e) y from 0 to 16 km and estimated it to be 0.05 from 0 to 80 km and 0.09 man rad per MW(e) y overall. The estimates include a shielding and occupancy factor of 0.5 and a tissue screening and absorption factor of 0.8. The collective dose estimates of Martin and the example in table 17 have not been corrected for shielding by buildings.

98. The main contributor to the collective dose from noble gases released from a PWR (table 17) is expected to be primarily ^{133}Xe, due to the long hold-up of the primary gaseous-waste processing system, but leakages also result in some release of shorter-lived noble gas nuclides. The collective dose appears to be primarily due to ^{133}Xe, ^{135}Xe and ^{88}Kr, with ^{133}Xe becoming increasingly a more important contributor to the dose at greater distances. Small releases of the longer-lived ^{133m}Xe (2.3 d) and ^{133m}Xe (12.0 d) occur, but their contributions to the collective dose remain small out to 1000 km. The contribution from ^{85}Kr (10.76 y) to the collective dose in the local and regional areas is insignificant.

99. The collective dose per unit of noble gas activity released by PWRs is $0.9\ 10^{-4}$ man rad Ci^{-1} within 100 km and $0.6\ 10^{-4}$ man rad Ci^{-1} from 100 to 1000 km (table 17). For the average PWR normalized release based on operating experience, about 15 Ci per MW(e) y (table 8), the local collective dose would be 0.0014 man rad per MW(e) y (to 100 km), with an additional regional contribution of 0.0011 man rad per MW(e) y (100-1500 km), including the extrapolated contribution from ^{133}Xe beyond 1000 km. The total of these two contributions is 0.0025 man rad per MW(e) y. Using actual population data and reported releases for two United States PWRs for 1972, Martin computed the collective dose to the population within 80 km of the reactors to be 0.005 man rad per MW(e) y (94).

2. Activation gases

100. Due to the short half-life of ^{41}Ar (1.83 h), the dose decreases quite rapidly with distance. A calculation of the individual gonad dose and collective dose is given in table 17 for the GCR, assuming the noble gas release to be 100 per cent ^{41}Ar. For a population density of $100\ \mathrm{km}^{-2}$, the collective dose per unit activity released to 100 km is $4.8\ 10^{-4}$ man rad Ci^{-1}. For an ^{41}Ar release from GCRs of 200 Ci per MW(e) y, the collective dose is estimated to be 0.1 man rad per MW(e) y. Clarke has calculated collective doses from ^{41}Ar released from Magnox reactors in the United Kingdom (24). His results are $(4.8\text{-}30)\ 10^{-4}$ man rad Ci^{-1}, depending on the reactor site, corresponding on the average to 0.13 man rad per MW(e) y. Reductions by factors of two to three would be appropriate to account for indoor shielding. Nagel has discussed initial attempts to compare calculations with measurements of the gamma dose from ^{41}Ar (103).

101. The behaviour in the environment of ^{35}S released from GCRs is not well understood. Ingestion of milk is thought to be the significant pathway to man; there are, however, uncertainties in estimating the deposition velocity and the transfer to milk. Passant has given a value of 0.023 rad per curie released as a rough estimate of the dose to bone marrow per unit activity released at the reactor boundary (113). Vennart and Ash have recently reviewed the dosimetry of inhaled and ingested ^{35}S (168).

102. The absorbed dose rate at the site boundary from ^{16}N released from GCRs is given as $5\ 10^{-5}$ rad y^{-1} (whole-body) for a 2000 Ci y^{-1} release (113). Absorbed dose rates in air from ^{16}N radiation originating from the turbine buildings of BWRs (about 5 mrad y^{-1} at 0.5 km) are quite dependent on source and field geometry (86, 87).

3. Tritium

103. Tritium present in airborne effluents from reactors can contribute to the local collective dose primarily through the inhalation pathway and secondarily through the ingestion pathway. The local collective dose contribution after deposition through the ingestion pathway may be quite variable from site to site owing to differences in local hydrology and water usage. This pathway is accounted for in the general assessment of the global collective dose commitment in chapter VI. The intake of airborne tritium occurs by inhalation (20 m^3 air inhaled per day) and by passage through skin. Intake through the skin is assumed to be equal to that through inhalation. The dose to soft tissue per unit activity intake in the form of tritiated water is 84 rad Ci^{-1} (12). Assuming an atmospheric dispersion factor of $5\ 10^{-7}\ \mathrm{s\,m}^{-3}$ at 1 km from the release and a reduction in concentration due to further dispersion inversely proportional to the 1.5 power of the distance expressed in kilometres, the local collective dose per unit activity released can be assessed by integration over the local area. Integrating from 1 to 100 km for a population density of $100\ \mathrm{km}^{-2}$ and including the above factors, we obtain the estimated local collective dose from airborne tritium per unit activity released:

$$S = (5\ 10^{-7}\ \mathrm{s\,m^{-3}})(40\ \mathrm{m^3\,d^{-1}})\left(\frac{1}{86\,400}\ \mathrm{d\,s^{-1}}\right)$$

$$(84\ \mathrm{rad\,Ci^{-1}})(100\ \mathrm{man\,km^{-2}})\int_{1}^{100} r^{-1.5}\, 2\pi r\, dr$$

the local collective dose from airborne tritium per unit activity released is estimated to be $2\ 10^{-4}$ man rad per Ci.

104. The average discharges of airborne tritium based on operating experience (table 11) are (Ci per MW(e) y), for PWRs and GCRs, 0.2; for BWRs, 0.05; for HWRs, 18. Using the above value of collective dose per unit release,

the estimated local collective doses, for the assumed local situation, are 4 10^{-5}, 1 10^{-5}, and 4 10^{-3} man rad per MW(e) y for the PWR and GCR, the BWR, and the HWR, respectively.

105. Tritium in liquid effluents from reactors can contribute most directly to the local collective dose if the discharge is to a river that provides a drinking water supply. The collective dose per unit activity released, estimated from the drinking water intakes given in table 22 (discussed in paragraph 126) is 0.007 man rad Ci^{-1}. The tritium releases reported from the most recent operating experience (table 12) are about 1 and 0.1 Ci per MW(e) y from PWRs and BWRs, respectively. From HWRs the release is assumed to be 4 Ci per MW(e) y. The local collective dose estimates would thus be 0.007, 0.0007, and 0.03 man rad per MW(e) y for the PWR, BWR, and HWR, respectively.

4. Carbon-14

106. Carbon-14 makes a very small contribution to the local collective dose. It is of more significance in terms of entering the carbon cycle and contributing to the global collective dose, as discussed in chapter VI. The local collective dose due to ^{14}C airborne releases from reactors can be estimated by comparing the resulting specific activity in air (which contains 0.16 g of C per cubic metre) to the natural specific activity (and thus the natural ^{14}C dose rate). Using the same atmospheric dispersion model and the same population assumptions as in the case of airborne tritium (para. 103), the local collective dose commitments per unit activity released are estimated to be 0.12, 0.1, 0.37, and 0.40 man rad Ci^{-1} for the lung, gonads, bone lining cells and bone marrow, respectively.

107. For a normalized ^{14}C release from a light-water reactor of 6 10^{-3} Ci per MW(e) y (para. 76), the local collective dose commitments are estimated to be 7 10^{-4}, 6 10^{-4}, 2.2 10^{-3} and 2.4 10^{-3} man rad per MW(e) y for the lungs, gonads, bone lining cells and bone marrow, respectively. As the inhalation intake of carbon is about 3 g d^{-1} and only 1 per cent of this intake is retained in the body (13), whereas the ingestion intake is 300 g d^{-1} with 100 per cent retention (68), the inhalation pathway can account for only 10^{-4} of the local collective dose. Regional contributions to the collective dose have been assessed by Kelly *et al.* (74), considering the first passage of the released material over a country or region. These contributions are also very small compared to the global collective dose commitment from ^{14}C.

5. Iodine

108. Assessment of local dose from ^{131}I released from reactors is complicated by the various forms that iodine may take—particulate, elemental, organic or as hypoiodous acid. Elemental iodine readily deposits on forage and enters the cow-milk-man pathway. Organic iodine, however, is retained much less efficiently by vegetation, and its deposition velocity is 200-1000 times smaller than that of the elemental form (2, 47). Particulate-associated iodine will be deposited at rates intermediate between those for the elemental and organic forms. The behaviour of hypoiodous acid is uncertain; it may simply decompose to the elemental and organic forms (114). Physico-chemical transformations occurring during atmospheric transport may also affect the amount of iodine in its various organic forms, since these are not stable in sunlight. The elemental form would be expected to become rapidly associated with airborne aerosols, so that deposition at distances beyond the immediate vicinity of the release would be largely governed by the particulate behaviour. This is also shown by fallout measurements (116).

109. A recent reassessment by Hoffman (53) of iodine deposition factors for predicting environment transfer has shown that the experimentally derived values of the deposition velocity have frequently been misinterpreted. As the measurements relate directly the actual deposition rate on vegetation to the concentration in air, the use of an additional factor to account for the intercepted fraction of the total deposition is therefore inappropriate.

110. Recent determinations of deposition velocities as the ratio of the total deposition on vegetation per unit ground surface area to the time-integrated concentration in air give 2 cm s^{-1} for the deposition of elemental iodine on grass and 0.01 cm s^{-1} or less for methyl iodide (47). In the case of particulates (mean aerosol particle diameter 4 μm), a value of 0.1 cm s^{-1} was found representative for grass, about 0.2 cm s^{-1} for clover and about 0.03 cm s^{-1} for vegetation-free soil (54). Since the deposition velocity varies with the vegetation density, less variability is encountered by normalizing the values by the mass of dry vegetation per unit area (52). For the conditions under which the particulate deposition velocities were measured, the normalized value for grass is 0.005 m^3 kg^{-1} s^{-1} (54).

111. The assessment of the transfer of iodine to milk, which is the main pathway to man, requires the knowledge of the value of the following parameters in addition to the deposition velocity: the residence half-time of particulates on vegetation (14 d), corresponding to an effective mean-life of 20.2 d for ^{129}I and 7.36 d for ^{131}I; the average mass of grass consumed per cow and per day under average agricultural conditions during the grazing season, taken to be 11 kg (dry) per day (51); the fractional transfer of the daily ingested activity by the cow per unit volume of produced milk, 0.01 d l^{-1}; and the fractional pasture-grazing time, assumed to be 0.5 for ^{131}I and 1.0 for ^{129}I. Taking account of all these parameters, an integrated air concentration of 1 pCi d m^{-3} of ^{131}I or ^{129}I would thus result in an integrated milk concentration for ^{131}I of 175 pCi d l^{-1} and for ^{129}I of 960 pCi d l^{-1}. Transfer via wet deposition is usually insignificant over the course of a year. For specific times or locations, however, it may indeed be significant.

112. To assess the transfer of iodine from air to fresh leafy vegetables, the required parameters are: the deposition velocity and the residence time on the vegetation (same values as discussed above); the fractional removal by washing, 0.4 (157); the decay due to an average 7-d marketing delay, if applicable, 0.55 for ^{131}I (157); and a dry-to-wet vegetation weight ratio of 0.5. Time-integrated concentrations of 350 pCi d kg^{-1}

(fresh weight) for ^{131}I and 1740 pCi d kg^{-1} (fresh weight) for ^{129}I are calculated for the case of a time-integrated air concentration of 1 pCi d m^{-3}.

113. An estimate of the transfer of ^{129}I to beef can also be made. The parameters required are: the deposition velocity, 0.005 m^3 kg^{-1} s^{-1}; residence mean-time on grass, 20.2 d; grass consumption rate, 11 kg (dry) per day; fractional transfer of daily ingested activity per unit mass of meat, 0.003 d kg^{-1} (111); and the fractional pasture time, 1.0. The resulting transfer factor is 290 pCi d kg^{-1} per 1 pCi d m^{-3} of ^{129}I in air.

114. The estimated transfer factors from air to milk, leafy vegetables or meat have not been verified in the case of actual radioiodine releases from reactors. A careful attempt to measure ^{131}I in the environment around a reactor during normal operation has been reported (170). Low levels of ^{131}I in air were detected near three BWRs, but only a few samples of vegetation and milk showed ^{131}I activity above the detection limit. The transfer of ^{131}I from deposition on grass to milk has also been estimated from fallout field measurements (see Annex C). These estimates of the transfer factor are somewhat lower than the value given in paragraph 11, but it has been suggested that the availability of fallout ^{131}I might be reduced because of its physical form (140). Because of the variability of all of the intervening factors, the assessment of transfer factor given in the preceding paragraphs can be expected to give only a general indication of the doses from airborne releases of iodine.

115. In a review of the iodine metabolic parameters certain values have been suggested for use in computing the age-dependent doses to the thyroid gland following an ingestion or inhalation intake (51). The values for ^{131}I and ^{129}I are given in table 18.

TABLE 18. DATA RELATING TO THE AGE-DEPENDENT DOSE TO THE THYROID GLAND FROM ^{131}I AND ^{129}I INTAKES

	Age			
	6 months	*4 years*	*14 years*	*Adult*
Thyroid mass (g)	2.0	4.0	14	20
Fractional uptake by the thyroid				
Ingestion	0.40	0.35	0.35	0.35
Inhalation	0.30	0.26	0.26	0.26
^{131}I				
Energy per disintegration (MEV)	0.18	0.18	0.19	0.19
Effective half-time in the thyroid (d)	6.0	6.3	6.9	7.6
Dose per unit intake (rad μCi^{-1})				
Ingestion	16	7.3	2.4	1.9
Inhalation	12	5.5	1.8	1.4
^{129}I				
Energy per disintegration (MEV)	0.060	0.061	0.063	0.064
Effective half-time in the thyroid (d)	23	28	48	136
Dose per unit intake (rad μCi^{-1})				
Ingestion	20	11	5.6	1.1
Inhalation	15	8.3	4.2	8.4

Source: Reference 51.

116. The average consumption rate of milk (68), leafy vegetables (157) and beef (one third of maximum meat consumption) (132) and the inhalation rate (68) for representative ages are given in table 19. These intake rates and the transfer factors estimated for the various pathways were used to compute the thyroid doses per unit time-integrated activity concentration in air, also listed in table 19. The highest dose is to the infant and is due almost entirely to milk consumption.

TABLE 19. DATA RELATING TO THE ENVIRONMENTAL PATHWAYS AND THYROID DOSES FROM ^{131}I and ^{129}I

	Age			
	6 months	*4 years*	*14 years*	*Adult*
Inhalation pathway				
Breathing rate (m^3 y^{-1})	1 150	3 530	6 440	8 030
Ingestion pathway consumption rate				
Milk (l/d)	0.9	0.5	0.4	0.25
Leafy vegetables (kg/y)	0	13	20	30
Beef (kg/y)	0	8	15	27
Dose per unit integrated concentration in air (mrad per pCi y m^{-3})				
^{131}I				
Inhalation	14	19	12	11
Ingestion[a]				
Milk	920	230	61	30
Leafy vegetables	0	33	17	20
Total[b]	930	280	90	61
^{129}I				
Inhalation	17	29	27	67
Ingestion[a]				
Milk	6 300	1 900	780	960
Leafy vegetables	0	250	190	570
Beef	0	26	24	86
Total[b]	6 300	2 200	1 000	1 700

[a] Particulate form assumed. Environmental transfer and therefore dose ~ 20 times higher for the elemental form and ~ 20 times less for the organic form.

[b] Rounded.

117. A population-weighted dose factor can be obtained by assuming that the doses to the three groups of children are representative of the age groups 0-1, 1-9, and 10-19 y, respectively, and that these groups contain respectively 2,16 and 20 per cent of the population (157), the population-weighted thyroid dose factor is 120 mrad per pCi y m^{-3} for ^{131}I and 1700 mrad per pCi y m^{-3} for ^{129}I.

118. The local thyroid collective dose commitment per unit activity of ^{131}I released from the reactor can be calculated using the same dispersion assumptions and population density values discussed previously:

$$S = (5\ 10^{-7}\,\mathrm{s\,m^{-3}})\left(\frac{1}{3.15\ 10^{7}}\,\mathrm{y\,s^{-1}}\right)$$

$$(120\ \mathrm{mrad\ pCi^{-1}\ y^{-1}\ m^{3}})\,(100\ \mathrm{man\ km^{-2}})$$

$$\int_{1}^{100} r^{-1.5}\,2\pi r\,dr$$

$$= 22\ \text{man rad per Ci}$$

On the basis of average operating experience (table 13), the normalized airborne release of ^{131}I from BWRs is $5\ 10^{-3}$ Ci per MW(e) y. The thyroid collective dose commitment per unit of electrical energy generated is therefore estimated to be 0.11 man rad per MW(e) y. The normalized collective dose commitments for PWRs, GCRs and HWRs are smaller by two orders of magnitude.

6. Particulates in airborne effluents

119. The activities of particulates released in airborne effluents may be quite variable but are relatively low. From releases based on average operating experience (table 14), estimates may be made of the local collective dose commitments from immersion and from inhalation intake while the radioactive material remains airborne and from the ingestion pathway and the exposure to contaminated ground following deposition.

120. The highest particulate activity discharge from PWRs was 0.0004 Ci per MW(e) y of ^{88}Rb. This would correspond to 0.004 Ci per MW(e) y of the precursor ^{88}Kr. Compared to ^{88}Kr activity releases of 280 Ci per MW(e) y from BWRs and 0.09 Ci per MW(e) y at PWRs, the release amounts and therefore the collective doses from ^{88}Rb particulate discharge are one or more orders of magnitude lower than those from ^{88}Rb that arise from discharge of ^{88}Kr.

121. Estimates of the collective dose commitments by the various pathways following unit release of activity in airborne effluents, are listed in table 20. The external gamma dose from the passing cloud (immersion exposure) makes a minor contribution to the collective dose commitment. The dose assessments from immersion and from inhalation assume decreasing concentrations with distance integrated to 100 km and a population density of 100 man km^{-2} in the local region. The dose estimates from the ingestion pathway and from exposure to contaminated ground do not require the deposition regions to be specified, provided that the population density can be assumed to be constant in the relevant area (100 km^{-2} is assumed). The transfer factors from deposition to diet and to man for strontium and caesium nuclides are derived from fallout field studies (see Annex C). The assessment of the exposure to contaminated ground is also discussed in Annex C for most of the radionuclides in table 20. The additional dose factors for ^{60}Co and ^{134}Cs assume a plane-source distribution maintained in time (10) and may thus be slight overestimates, as some downward movement in soil is expected over the mean-lives of these radionuclides.

TABLE 20. LOCAL COLLECTIVE DOSE COMMITMENT FACTORS FOR RADIONUCLIDES IN AIRBORNE PARTICULATE EFFLUENTS

	Collective dose commitment per unit activity released (man rad Ci^{-1})						
		Inhalation		*Ingestion*			*Exposure to contaminated ground*
Radionuclide	*Air immersion total body*	*Lung (whole)*	*Bone marrow*	*Total body*	*Bone marrow*	*Bone lining cells*	*Total body*
^{58}Co	0.001	0.09	0.005				1.2
^{60}Co	0.003	1.9	0.08				80.0
^{89}Sr		0.01	0.02		0.06		
^{90}Sr		0.03	1.0		100	140	
^{106}Ru	0.0002	5.6	0.009				1.9
^{134}Cs	0.002	0.07	0.07	30			21.0
^{137}Cs	0.0007	0.06	0.05	20			46.0
^{140}Ba	0.0003	0.009	0.005				0.5

122. Estimate of the collective dose commitments due to radioactive particulates released in airborne effluents from reactors per unit electrical energy generated are given in table 21, obtained by applying the dose factors of table 20 to the releases representing average operating experience. The collective dose commitments are somewhat higher for BWRs than PWRs, ranging from $3\ 10^{-5}$ to $2\ 10^{-4}$ man rad per MW(e) y to the lung, due primarily to ^{60}Co, and $3\ 10^{-3}$ to $1\ 10^{-2}$ man rad per MW(e) y to total body, due to ^{60}Co from external exposure and to ^{134}Cs and ^{137}Cs from external exposure and through the ingestion pathway. The dose commitments to bone marrow and bone lining cells are due almost entirely to the small amounts of ^{90}Sr released.

TABLE 21. LOCAL COLLECTIVE DOSE COMMITMENTS FROM PARTICULATES IN AIRBORNE EFFLUENTS FROM REACTORS

	Normalized activity released (Ci per MW(e) y)	*Collective dose per unit electrical energy generated (man rad per MW(e) y)*			
		Total body	*Lung (whole)*	*Bone marrow*	*Bone lining cells*
P W R					
^{137}Cs	$2.1\ 10^{-5}$	$1.4\ 10^{-3}$	$0.1\ 10^{-5}$	$0.1\ 10^{-5}$	
^{60}Co	$1.5\ 10^{-5}$	$1.2\ 10^{-3}$	$2.9\ 10^{-5}$	$0.1\ 10^{-5}$	
^{134}Cs	$1.4\ 10^{-5}$	$0.7\ 10^{-3}$	$0.1\ 10^{-5}$	$0.1\ 10^{-5}$	
^{58}Co	$1.6\ 10^{-5}$	$0.02\ 10^{-3}$	$0.1\ 10^{-5}$	$0.08\ 10^{-6}$	
^{140}Ba	$3.2\ 10^{-7}$	$0.02\ 10^{-5}$	$0.03\ 10^{-7}$	$0.02\ 10^{-7}$	
^{90}Sr	$1.4\ 10^{-7}$		$0.04\ 10^{-7}$	$1.4\ 10^{-5}$	$2.0\ 10^{-5}$
^{89}Sr	$6.0\ 10^{-7}$		$0.06\ 10^{-7}$	$0.05\ 10^{-6}$	
	Total	$3\ 10^{-3}$	$3\ 10^{-5}$	$2\ 10^{-5}$	$2\ 10^{-5}$
B W R					
^{60}Co	$1.0\ 10^{-4}$	$8.0\ 10^{-3}$	$1.9\ 10^{-4}$	$0.08\ 10^{-4}$	
^{134}Cs	$2.3\ 10^{-5}$	$1.2\ 10^{-3}$	$0.02\ 10^{-4}$	$0.02\ 10^{-4}$	
^{137}Cs	$1.7\ 10^{-5}$	$1.1\ 10^{-3}$	$0.01\ 10^{-4}$	$0.09\ 10^{-5}$	
^{140}Ba	$3.9\ 10^{-4}$	$0.2\ 10^{-3}$	$0.04\ 10^{-4}$	$0.02\ 10^{-4}$	
^{58}Co	$1.8\ 10^{-5}$	$0.02\ 10^{-3}$	$0.02\ 10^{-4}$	$0.09\ 10^{-6}$	
^{90}Sr	$5.0\ 10^{-6}$		$0.02\ 10^{-5}$	$5.1\ 10^{-4}$	$7.0\ 10^{-4}$
^{89}Sr	$3.3\ 10^{-4}$		$0.03\ 10^{-4}$	$0.3\ 10^{-4}$	
	Total	$1\ 10^{-2}$	$2\ 10^{-4}$	$5\ 10^{-4}$	$7\ 10^{-4}$

7. Radionuclides in liquid effluents

123. Radionuclides discharged in liquid effluents may result in doses to man through the pathways of drinking water and fish consumption for releases to fresh water, and of ocean fish and shellfish consumption for releases to salt water. A portion of the population may also be exposed on shorelines to external irradiation from sediments. In areas where irrigation of crops is practised, especially when sprinkling methods are utilized, a significant contribution to the collective dose could result. This pathway should then also be considered. Ingestion of water fowl and immersion exposure from swimming are other pathways, but these have generally been found to be insignificant contributors to the collective dose (19).

124. Assessments for specific reactors can be performed by means of rather specific transfer models and using the information provided by environmental monitoring. This monitoring is usually carried out along the critical pathways leading to human exposures. Site-specific assessments for the liquid discharges of nine GCRs in the United Kingdom (106) show that the local annual collective dose to the gonads in the vicinity of these reactors is in the range of 10^{-1} to 1 man rad. Combining these values with the electrical energy generated in each case (106), the normalized gonad collective dose is estimated to be in the range of $5\ 10^{-4}$ to $4\ 10^{-3}$ man rad per MW(e) y.

125. In the general case, the dose assessments require information on the discharged activity of each radionuclide, the volume of the receiving water into which the radioactivity is diluted, the concentration levels which are reached in fish and shellfish, the factors regarding removal to sediments and exchange rates of water bodies, and the number of individuals in the populations who use the water for drinking purposes and who consume fish.

126. The local and regional collective dose commitments from radionuclides in liquid effluents can be estimated using the expression

$$S_i^c = \frac{A_i}{V(\tau + \lambda_i)} \sum_k N_k f_{k,i} I_k D_i$$

where

A_i (pCi) = activity of nuclide i released to water

V (l) = volume of receiving water

τ (y^{-1}) = turnover time of receiving water

λ_i (y^{-1}) = radioactive decay constant for nuclide i

N_k (man) = number of individuals for pathway k

$f_{k,i}$ (pCi kg^{-1} per pCi l^{-1}) = concentration factor for item in pathway k for nuclide i

I_k (kg y^{-1}) = individual consumption rate of pathway item k

D_i (rad Ci^{-1}) = tissue dose per unit activity ingested.

By using the normalized release rate for A_i (Ci per MW(e) y), the collective dose will be expressed in units of man rads per MW(e) y. The total collective dose commitment per unit of electrical energy generated S^c is obtained by a further summation over all radionuclides.

127. The quantity $\frac{A}{V(\tau + \lambda)}$ (pCi y l^{-1}) is the integral water concentration for release of an activity A (pCi) or, alternatively, the equilibrium concentration in water $C_{w,i}$ (pCi l^{-1}), for a constant continuing release rate (Ci y^{-1}). The equilibrium concentration in fish or shellfish is $C_{w,i} f_{k,i}$ (pCi/kg fish), where $f_{k,i}$ is the appropriate fresh water or salt water concentration factor.

128. For radionuclide releases to small volumes of water, the concentrations in water or fish may be high, but the population which can be served with drinking water or by fish production will be limited. For releases to larger water volumes, the concentrations will be less, but the populations involved will be correspondingly larger. In fact, the N_k/V relationship could be taken, in a crude approximation, as relatively constant, the inverse of which indicates the water utilization with regard to the specific pathway k per individual in the population.

129. For the drinking water pathway, a fresh water utilization ratio of $3\ 10^7$ l man^{-1} will be assumed in estimating collective dose commitments from generalized liquid releases. This value of V/N_k is assumed to be a global average, obtained from an estimated global total of $1.3\ 10^{17}$ l of fresh water in lakes ($1\ 10^{17}$ l) and rivers (annual flow $0.3\ 10^{17}$ l) (5), serving the world population of $3.8\ 10^9$. Similar results are obtained for regional areas. Martin (93) estimates that a population of $3.1\ 10^6$ derives drinking water from Lake Ontario and the St. Lawrence River, comprising an available volume of $2.2\ 10^{14}$ l. This gives a water utilization factor of $7\ 10^7$ l man^{-1}. In the Danube River studies, Ruf (122) quotes annual flows of (3-7) 10^{12} l, from which several million derive their drinking water. This gives a factor of about $2\ 10^6$ l man^{-1}. For the large flow of the Mississippi River, the estimated factor is $3\ 10^8$ l man^{-1} (159). Even more widely varying results would be possible for more localized situations.

130. Average fish plus sea-food consumption per individual is about 8 kg y^{-1}, ranging from 4 to 6 kg y^{-1} in the Near East and Africa to 10-14 kg y^{-1} in the Far East and Europe (68). It may be assumed that the annual consumption is 6 kg y^{-1} ocean fish, 1 kg y^{-1} fresh water fish and 1 kg y^{-1} shellfish. Total fresh water fish consumption by the world population is thus $3.8\ 10^9$ kg y^{-1}, which, when a correction is made for edible weight, agrees well with the estimated annual global harvest of 10^{10} kg landed weight (5). Dividing by the global fresh water volume given in the above paragraph, we obtain $3\ 10^{-8}$ man kg y^{-1} l^{-1}, which will be assumed to be the factor $N_k I_k/V$ needed in estimating collective doses from fresh water fish consumption.

131. Similarly, ocean-fish consumption by the world population is 22.8 10^9 kg y^{-1}; The annual global ocean fish harvest is estimated to be 60 10^9 kg landed weight (5). The ocean surface area is 3.6 10^8 km^2, and assuming the depth for fish production extends to 30 m, the water volume is 10^{19} l. Of course, most of the fish are taken from coastal regions, comprising only about 10 per cent of the ocean's surface. However, most of the larger water volume can by expected to circulate through the fish production regions. Thus, the factor $N_k I_k / V$ which is required for the salt water fish pathway is 2 10^{-9} man kg y^{-1} l^{-1}. For the shellfish pathway, with consumption one sixth that of ocean fish, a value of the factor $N_k I_k / V$ of 3 10^{-10} man kg y^{-1} l^{-1} will be used in this Annex.

132. For estimating collective doses from exposure to sediments, rather more specific assumptions must be made regarding the receiving waters and the potential population involved. Booth *et al.* (19) have calculated sediment exposure rates based on releases to a water volume of 10^{12} l with a water turnover rate of 0.1 y^{-1}. This corresponds to an annual dilution volume of 10^{11} l. Dividing by the fresh water utilization factor derived above of 3 10^7 l man^{-1}, it can be assumed that the exposures apply to a population of 3000. These values will be used to obtain collective dose estimates for sediment exposure in the general release situation.

133. For assessments involving releases to the sea, the dilution volume will be assumed to be 10^{15} l, with a turnover of 1 y^{-1}. The salt water utilization factor from global values of ocean volume and population is 10^{19} l per 3.8 10^9 man, or 3 10^9 l man^{-1}. Thus, for a dilution volume of 10^{15} l, the corresponding population is 300 000. For releases to salt water, this value will be used to obtain collective doses from the individual dose rates from sediment exposure calculated by Booth *et al.* (19). The estimates of populations exposed to sediments are admittedly rather arbitrarily selected and should certainly be adjusted for estimating the collective dose at specific sites.

134. Correction factors to account for removal of radionuclides from water to sediments depend on the turnover rate, that is, the flow-through or replacement rate of the receiving water. The correction becomes less important for fast moving water ($\tau = 10$ or 100 y^{-1}). For releases to salt water, in which depths greater than 10 m can be assumed, the correction is also negligible (19). This, however, will depend on how the effluents are admitted to the water. For near-shore releases not far from the bottom, sediment removal is considerable for many radionuclides. Even for surface releases, sedimentation may in some cases be an important process, ultimately removing the activity from the water. Whether sediments once deposited might eventually become resuspended by storms or by winds from shore areas is a question that is being studied.

135. In the case of releases to fresh water, a turnover rate of 0.1 y^{-1} will be assumed for the generalized assessments. For ^{137}Cs, the correction for sediment removal is taken to be 0.29 (19); therefore, the time-integrated activity for an activity release of 1 Ci is $0.29(\tau + \lambda)^{-1} = 2.4$ Ci y. For Lake Huron, which was analysed by Barry (4), the integrated value for a unit activity release of ^{137}Cs was 1.4 Ci y, in reasonable agreement with the assumed generalized case. Mitchell and Jeffries (101), however, estimate a more rapid turnover rate for Lake Trawsfynydd, the half-time being 50 d, corresponding to $\tau = 5$ y^{-1}. For releases to salt water, a turnover rate of 1 y^{-1} will be assumed with no additional removal of activity to sediments. Kautsky (73) estimates the turnover of water in the North Sea to be of this order, and Hetherington (48) reports the residence half-time of water in the Irish Sea to be approximately 1 y.

136. The factors which are required to estimate collective doses (using the formula in paragraph 126) for a number of radionuclides that may be present in liquid effluents released to fresh and salt water are listed in table 22. The corrections to account for removal to sediments are also included, and in addition, factors are given for the removal of activity during treatment of drinking water (30, 122). The concentration factors for fresh-water fish are those compiled by Thompson (135) and for marine fish and shellfish by Freke (35). The factors relating dose to activity intake included in table 22 have the follow characteristics: (a) values for the total body (TB) from ^{3}H, ^{14}C, ^{65}Zn, ^{134}Cs, and ^{137}Cs apply approximately equally to most single-organ exposures; (b) the dose factors for thyroid (Thy) are taken from table 18; (c) for ^{90}Sr the dose factor used refers to exposure of bone lining cells; the dose to bone marrow would be 30 per cent smaller; (d) the doses from ^{32}P, ^{89}Sr, and ^{239}Pu are in total bone; (e) for the other radionuclides the highest doses are experienced in the lower large intestine (LLI).

137. Collective dose estimates for the composition and amounts of radionuclides other than ^{3}H in liquid effluents from reactors reported from operating experience (table 15) are given in table 23. The collective doses to the total body include contributions from ingestion of ^{137}Cs, ^{134}Cs and ^{65}Zn, and from external exposure of ^{137}Cs, ^{134}Cs, ^{60}Co, ^{54}Mn, ^{144}Ce, ^{51}Cr, ^{125}Sb, and ^{106}Ru in sediments. The dose to the lower large intestine is due primarily to ^{60}Co and ^{54}Mn for PWRs and BWRs and to ^{125}Sb for GCRs. The dose to bone from ^{32}P in releases to salt water from GCRs is estimated to be comparable to the ^{90}Sr dose, but is much less for release to fresh water. Since the average radionuclide release from GCRs applies chiefly to reactors which discharge to the sea, a separate example is included in table 23 for discharges to fresh water from the GCR at Lake Trawsfynydd.

138. The estimates of collective dose in table 23 correspond to average release amounts from reactors and for generalized release situations. The collective doses for specific reactor situations may differ strongly from them. There may also be particular radionuclides or special pathways which might be overlooked. In general, however, the results seem to be consistent with the dose estimates from environmental sampling (15, 70, 71, 100).

TABLE 22. COLLECTIVE DOSE FACTORS FOR RADIONUCLIDES IN LIQUID EFFLUENTS RELEASED TO FRESH AND SALT WATER

								Collective dose per unit activity released (man rad Ci^{-1})				
Radionuclide	*Radioactive half-life* $T_{1/2}$	*Sediment removal correction factor* s_i	*Time integral of unit activity in water* W_i *(Ci y)*	*Drinking-water treatment removal factor* f_{1i}	*Concentration factor, fish* f_{2i} *(pCi kg^{-1} fish per pCi l^{-1} water)*	*Dose per unit activity ingested* D_i *(10^{-6} rad Ci^{-1})*	*External dose from sediment per unit activity released (10^{-6} rad Ci^{-1})*	*Drinking water* S_{1i}	*Fish* S_{2i}	*Total (internal)*		*Sediment exposure (external)*
							Fresh water					
^{3}H	12.26 y	1.0	6.39	1.0	0.9	0.000084		0.0071	1.4 10^{-5}	0.0071	TB	
^{14}C	5 730 y	0.78	7.79	1.0	4 600	0.00056		0.057	0.60	0.66	TB	
^{32}P	14.3 d	1.0	0.056	0.4	100 000	0.038		0.011	6.4	6.4	Bone	
^{51}Cr	27.7 d	0.97	0.105	0.9	40	0.00097	0.0023	0.0012	0.00012	0.0013	LLI	6.8 10^{-6}
^{54}Mn	313 d	0.83	0.914	0.5	100	0.019	2.76	0.11	0.052	0.16	LLI	0.0083
^{60}Co	5.26 y	0.33	1.43	0.5	20	0.039	162	0.37	0.033	0.40	LLI	0.48
^{65}Zn	244 d	0.96	0.845	0.4	1 000	0.0065	0.30	0.029	0.16	0.19	TB	0.00090
^{89}Sr	50.5 d	1.0	0.196	0.5	5	0.062		0.080	0.0018	0.082	Bone	
^{90}Sr	28.5 y	0.78	6.27	0.5	5	0.7		29.0	0.66	30.0	Bone	
^{95}Zr	63 d	0.81	0.197	0.1	3.3	0.032	1.13	0.0083	0.00062	0.0089	LLI	0.0034
^{95}Nb	35 d	1.0	0.136	0.1	30 000	0.019	0.0090	0.0034	2.3	2.3	LLI	2.7 10^{-5}
^{103}Ru	41 d	0.92	0.147	0.1	10	0.024	0.11	0.0046	0.0011	0.0057	LLI	0.00032
^{106}Ru	1.0 y	0.59	0.743	0.1	10	0.19	1.73	0.19	0.042	0.23	LLI	0.0052
^{110m}Ag	270 d	1.0	0.964	0.5	2.3	0.065	0.84	0.41	0.0043	0.41	LLI	0.0025
^{125}Sb	2.7 y	0.94	2.63	0.5	1	0.019	2.15	0.33	0.0015	0.33	LLI	0.0065
^{129}I	1.6 10^{7} y	1.0	10.0	0.8	15	10.0		1 050	45	1 100	Thy	
^{131}I	8.05 d	1.0	0.032	0.8	15	3.1	6.7 10^{-5}	1.0	0.045	1.1	Thy	2.0 10^{-7}
^{134}Cs	2.1 y	0.51	1.19	0.2	400	0.075	36	0.23	1.1	1.3	TB	0.11
^{137}Cs	30.0 y	0.29	2.36	0.2	400	0.043	235	0.27	1.2	1.5	TB	0.70
^{144}Ce	290 d	0.63	0.648	0.1	25	0.19	0.33	0.16	0.092	0.25	LLI	0.00099
^{239}Pu	24 400 y	1.0	10.0	0.1	10	0.019		0.25	0.06	0.31	Bone	
							Salt water					
				Concentration factor				*Concentration factor*				
				Fish C_{3i}	*Shellfish* C_{4i}			*Fish* S_{3i}	*Shellfish* S_{4i}			
^{3}H	12.26 y	1.0	0.946	1	1	0.000084		1.6 10^{-7}	2.4 10^{-8}	1.8 10^{-7}	TB	
^{14}C	5 730 y		1.0	1	1	0.00056		1.1 10^{-6}	1.7 10^{-7}	1.3 10^{-6}	TB	
^{32}P	14.3 d		0.056	10 000	10 000	0.038		0.043	0.0064	0.049	Bone	
^{51}Cr	27.7 d		0.099	100	1 000	0.00097	1.1 10^{-6}	1.9 10^{-5}	2.9 10^{-5}	4.8 10^{-5}	LLI	3.3 10^{-7}
^{54}Mn	313 d		0.553	3 000	50 000	0.019	0.0006	0.063	0.16	0.22	LLI	0.00018
^{60}Co	5.26 y		0.884	100	10 000	0.039	0.0052	0.0069	0.10	0.11	LLI	0.0016
^{65}Zn	244 d		0.491	5 000	50 000	0.065	0.00016	0.032	0.048	0.082	TB	4.8 10^{-5}
^{89}Sr	50.5 d		0.166	1	1	0.062		2.1 10^{-5}	3.1 10^{-6}	2.4 10^{-5}	Bone	
^{90}Sr	28.5 y		0.976	1	1	0.7		0.0014	0.00020	0.0016	Bone	
^{95}Zr	63 d		0.199	30	100	0.032	0.0042	0.00038	0.00019	0.00057	LLI	0.0013
^{95}Nb	35 d		0.122	100	200	0.019	0.00004	0.00046	0.00014	0.00060	LLI	1.2 10^{-6}
^{103}Ru	41 d		0.139	3	100	0.024	0.000012	0.00002	0.0001	0.00012	LLI	3.6 10^{-6}
^{106}Ru	1.0 y		0.591	3	100	0.19	0.000076	0.00067	0.0034	0.0041	LLI	2.3 10^{-5}
^{110m}Ag	270 d		0.516	1 000	5 000	0.065	0.00074	0.067	0.050	0.12	LLI	0.00022
^{125}Sb	2.7 y		0.796	1 000	1 000	0.019	0.00025	0.030	0.0045	0.035	LLI	7.5 10^{-5}
^{129}I	1.6 10^{7} y		1.0	20	100	10.0		0.40	0.30	0.70	Thy	
^{131}I	8.05 d		0.031	20	100	3.1	3.6 10^{-7}	0.0038	0.0029	0.0067	Thy	1.1 10^{-7}
^{134}Cs	2.1 y		0.752	30	50	0.075	0.0016	0.0034	0.00085	0.0043	TB	0.00049
^{137}Cs	30.0 y		0.977	30	50	0.043	0.0082	0.0025	0.00063	0.0031	TB	0.0025
^{144}Ce	290 d		0.534	30	100	0.19	0.0028	0.0061	0.0030	0.0091	LLI	0.00084
^{239}Pu	24 400 y		1.0	3	300	0.019		0.00011	0.0017	0.0018	Bone	

Notes: i = radionuclide; k = pathway = 1, 2, 3, 4 for drinking water, fresh-water fish, salt-water fish and shellfish, respectively; λ_i = decay constant = $0.693/T_{1/2}$; τ = water turnover rate = 0.1 y^{-1} (fresh water), 1.0 y^{-1} (salt water); $S_{k,i} = W_i f_{k,i} N_k I_k D_i / V_i$; where $W_i = A_i s_i (\tau + \lambda_i)^{-1}$; $I_1 = 438\ l\ y^{-1}$, $N_1/V = 3\ 10^{-8}$ man l^{-1}; $I_2 = 1$ kg y^{-1}, $N_2 I_2 / V = 3\ 10^{-8}$ man kg $y^{-1}\ l^{-1}$; $I_3 = 6$ kg y^{-1}, $N_3 I_3 / V = 2\ 10^{-9}$ man kg $y^{-1}\ l^{-1}$; $N_4 I_4 / V = 3\ 10^{-10}$ man kg $y^{-1}\ l^{-1}$.

TABLE 23. COLLECTIVE DOSE FROM RADIONUCLIDES OTHER THAN TRITIUM IN LIQUID EFFLUENTS FROM REACTORS

			Normalized collective dose (man rad per MW(e) y)							
			Release to salt water				*Release to fresh water*			
	Normalized activity released (Ci per MW(e) y)		*Total body*	*Lower large intestine*	*Bone*	*Thyroid*	*Total body*	*Lower large intestine*	*Bone*	*Thyroid*
P W R										
^{137}Cs	0.002		1.1 10^{-5}				4.4 10^{-3}			
^{134}Cs	0.001		0.5 10^{-5}				1.4 10^{-3}			
^{60}Co	0.0004		0.06 10^{-5}	4.4 10^{-5}			0.2 10^{-3}	1.6 10^{-4}		
^{54}Mn	0.0001		0.02 10^{-6}	2.2 10^{-5}			0.08 10^{-5}	1.6 10^{-5}		
^{144}Ce	0.00008		0.07 10^{-6}	0.07 10^{-5}			0.08 10^{-6}	0.2 10^{-4}		
^{51}Cr	0.0002		0.01 10^{-8}	0.01 10^{-6}			0.01 10^{-7}	0.03 10^{-5}		
^{90}Sr	0.000002				3.2 10^{-9}				6.0 10^{-5}	
^{131}I	0.002					1.3 10^{-5}				2.2 10^{-3}
	Total		2 10^{-5}	7 10^{-5}	3 10^{-9}	1 10^{-5}	6 10^{-3}	2 10^{-4}	6 10^{-5}	2 10^{-3}
B W R										
^{137}Cs	0.025		1.4 10^{-4}				5.5 10^{-2}			
^{134}Cs	0.017		0.8 10^{-4}				2.4 10^{-2}			
^{60}Co	0.006		0.1 10^{-4}	6.6 10^{-4}			0.3 10^{-2}	2.4 10^{-3}		
^{54}Mn	0.002		0.04 10^{-5}	4.4 10^{-4}			0.02 10^{-3}	0.3 10^{-3}		
^{51}Cr	0.0006		0.02 10^{-8}	0.03 10^{-6}			0.04 10^{-7}	0.08 10^{-5}		
^{90}Sr	0.0001				1.6 10^{-7}				3.0 10^{-3}	
^{89}Sr	0.0008				0.2 10^{-7}				0.07 10^{-3}	
^{131}I	0.004					2.7 10^{-5}				4.4 10^{-3}
	Total		2 10^{-4}	1 10^{-3}	2 10^{-7}	3 10^{-5}	8 10^{-2}	3 10^{-3}	3 10^{-3}	4 · 10^{-3}
G C R	*Salt water*	*Fresh water*								
^{137}Cs	0.068	0.006	3.8 10^{-4}				1.3 10^{-2}			
^{134}Cs	0.014	0.0009	0.7 10^{-4}				0.1 10^{-2}			
^{65}Zn	0.00008	0.00005	0.06 10^{-4}							
^{60}Co	0.0007	0.0001	0.01 10^{-4}	0.8 10^{-4}						
^{125}Sb	0.008	0.008	0.08 10^{-5}	2.8 10^{-4}			0.05 10^{-3}	2.6 10^{-3}		
^{144}Ce	0.001	0.00005	0.08 10^{-5}	0.09 10^{-4}			0.05 10^{-6}	0.01 10^{-3}		
^{106}Ru	0.002	0.0008	0.05 10^{-6}	0.08 10^{-4}			0.04 10^{-4}	0.2 10^{-3}		
^{32}P	0.0008	0.0007			3.9 10^{-5}				0.4 10^{-2}	
^{90}Sr	0.01	0.0005			1.6 10^{-5}				1.5 10^{-2}	
^{89}Sr	0.0007				0.02 10^{-6}					
		Total	5 10^{-4}	4 10^{-4}	6 10^{-5}		1 10^{-2}	3 10^{-3}	2 10^{-2}	

IV. FUEL REPROCESSING

139. At the fuel reprocessing stage of a completely integrated nuclear fuel cycle, the uranium and plutonium in the irradiated fuel are recovered for reuse from the fission products that have accumulated during fuel burn-up in the reactor. The spent fuel received at the reprocessing plant is stored under water, which serves both as a transparent radiation shield and as a coolant. The fuel elements to be processed are transferred to a head-end cell, where they are disassembled, and the individual rods are treated mechanically to expose the fuel. The fuel is then leached with nitric acid

140. A solvent extraction process is then used to separate the uranium and plutonium from the fission and activation products. Nearly all fuel reprocessing facilities employ some form of the Purex process, which uses the organic complexing compound tributyl phosphate to extract both the uranium and the plutonium into the organic phase. The uranium and plutonium can be separately back-extracted from the organic solution with nitric acid. The product nitrates are further purified and converted to oxides suitable for refabrication into fuel elements.

A. EFFLUENTS

141. Waste processing to prevent large releases of radioactivity to the environment is a complex task at reprocessing plants. The gaseous and volatile fission products (I, ^3H, Kr, Xe, Ru, Te and Cs) are largely separated from the fuel during the fuel dissolution step. The dissolver off-gas is treated for nitric acid recovery and for iodine removal before being blended with the off-gas from other vessels in the plant. The vessel off-gas is usually treated by caustic scrubbing, drying and filtering through high efficiency filters before discharge from a stack. The aqueous wastes, containing almost all of the fission products, are generally concentrated by evaporation and stored in underground tanks and will eventually be solidified. The evaporator overheads are usually sufficiently decontaminated of radioisotopes to permit their discharge to the environment under existing regulations.

TABLE 24. RADIONUCLIDES DISCHARGED FROM VARIOUS FUEL REPROCESSING PLANTS, 1960-1974

(a) In airborne effluents

Year	*Electrical energy equivalent of fuel processed (MW(e) y)*	*Activity released (Ci)* ^{85}Kr	^{3}H	^{90}Sr	^{106}Ru	^{131}I	^{129}I	^{89}Sr	^{137}Cs	^{134}Cs
		Windscale (United Kingdom)								
1972	~ 2 500	1 000 000		0.097		23.6	0.004			
1973				0.75		1.2				
1974				0.15		0.01				
Total (Ci)		1 000 000		1.0		24.8	0.004			
Normalized release 1972 (Ci per MW(e) y)		400		4 10^{-5}		9 10^{-3}	2 10^{-6}			
		Nuclear Fuel Services (United States)								
1966	210	77 000								
1967	900	328 000	~ 500			< 0.06				
1968	530	193 000	~ 500			0.0066				
1969	830	300 000	~ 500			< 0.03				
1970	660	180 000	~ 500			< 0.03				
1971	720	221 000	1 070	0.00036	0.0013	0.0024	0.0026	0.00027	0.0009	0.0006
1972	0	0	5.5	0.041	0.029	0.00096		0.0021		
Total (Ci)		1 299 000	~3 075	0.041	0.030	0.01	0.0026	0.0024	0.0009	0.0006
Normalized release 1966-1972 (Ci per MW(e) y)		340	0.84	6 10^{-5}	4 10^{-5}	8 10^{-6}	4 10^{-6}	3 10^{-6}	1 10^{-6}	8 10^{-7}

(b) In liquid effluents

La Hague (France)

Year		^{106}Ru	^{137}Cs	^{144}Ce	^{110m}Ag	$^{95}Zr\text{-}Nb$
1971		7 900	6 700	3 900		90
1972		7 600	1 200	1 400		340
1973		7 000	2 000	1 700	1 450	70
	Total (Ci)	22 500	9 900	7 000	1 450	500

Windscale (United Kingdom)

Year		^{137}Cs	^{3}H	^{106}Ru	^{95}Zr	^{95}Nb	^{90}Sr	^{144}Ce	^{134}Cs	^{239}Pu	^{103}Ru	^{89}Sr
1960		910		39 600	2 400	6 300	520	890			11 600	980
1961		1 100		25 300	1 700	7 900	490	2 200			3 200	1 370
1962		1 100		23 000	940	4 300	1 020	2 400			1 800	500
1963		370		33 400	560	3 300	550	1 400			9 600	170
1964		1 300		23 100	21 600	20 800	970	3 200			1 200	190
1965		1 200		21 000	17 700	33 700	1 160	3 500			1 800	170
1966		1 200		24 900	14 100	23 400	910	6 900			2 500	90
1967		1 600	16 100	17 200	18 800	25 700	1 390	13 700		492	2 200	140
1968		1 500	20 300	24 200	28 100	37 100	1 370	10 000		828	1 800	40
1969		12 000	24 800	22 900	31 600	30 000	2 950	13 500	630	816	1 400	230
1970		30 100	32 000	27 600	9 100	9 900	6 000	12 400	7 010	936	890	470
1971		35 800	31 500	36 400	18 000	17 300	12 300	17 200	6 370	1 128	830	390
1972	≈ 2 500	34 800	33 600	30 500	25 600	23 500	15 200	13 700	5 820	1 558	1 160	1 080
1973				37 800			7 450	14 600		1 776		
1974				29 200			10 600	6 530		1 248		
Total (kCi)		123.0	158.3	416.1	190.2	243.2	62.9	122.0	19.8	8.8	40.0	5.8
Normalized release 1972 (Ci per MW(e) y)		·13.9	13.4	12.2	10.2	9.4	6.1	5.5	2.3	0.62	0.46	0.43

TABLE 24 *(continued)*

Year	*Electrical energy equivalent of fuel processed (MW(e) y)*	*Activity released (Ci)*								
		Nuclear Fuel Services (United States)								
		^{3}H	*^{106}Ru*	*^{137}Cs*	*^{90}Sr*	*^{89}Sr*	*^{134}Cs*	*^{129}I*	*^{239}Pu*	*^{238}Pu*
1966	210	290								
1967	900	4 200			9.9			0.07		
1968	530	2 650			5.0			0.03		
1969	830	5 980			10.1			0.2		
1970	660	4 520			14.2			0.34		
1971	720	3 850	48.2	10.0	6.6	2.27	1.83	0.21	0.0048	0.0042
1972	0	600	6.1	1.1	0.7	0.08	0.19	0.06	0.0005	0.0007
	Total (Ci)	22 090	54.3	11.1	46.5	3.35	2.02	0.91	0.0053	0.0049
Normalized release 1966-1972 (Ci per MW(e) y)		5.7	0.075	0.015	0.013	0.0033	0.0028	$2.5\ 10^{-4}$	$7.4\ 10^{-6}$	$6.8\ 10^{-6}$

Sources: References 21, 48, 50, 93, 100, 106, 123, 134, 147, 172.

142. The radionuclides of concern in reprocessing plant effluents include primarily ^{3}H, ^{14}C, ^{85}Kr, ^{129}I, ^{131}I, ^{106}Ru, ^{134}Cs, ^{137}Cs, and the actinides. General discussions of the occurrence and release pathways from reprocessing plants have appeared in several reports (8, 83, 107, 109, 151, 158), and more detailed accounts have been published for tritium (39, 80), noble gases (49, 80), and iodine (16, 118). Specific information available on radionuclide discharges from reprocessing plant operating experience is given in table 24. The release data from the NFS plant in the United States represent a completed experience, as this plant was closed for modifications in 1972.

143. The release of radioactive materials from the fuel reprocessing plant depends upon the type of fuel, irradiation history, cooling time and the specific waste processing systems of the reprocessing installation. The typical radionuclide composition of LWR fuel following an average burn-up of 33 000 MW(t) d per tonne and a cooling time of 150 d is given in table 25 (107). Burn-up differs for the various light-water reactor types. The burn-up levels achieved in presently operating reactors, and therefore the amounts of longer-lived radionuclides generated, have been somewhat less than the example given in table 25. The cooling time has also been on the average much longer than 150 d (167).

TABLE 25. RADIONUCLIDE CONTENT OF LWR FUEL

Burn-up, 33 000 MW(t) d t^{-1}; cooling time, 150 d

Radio-nuclide	*Per unit mass of fuel (Ci t^{-1})*	*Per unit electricity generated[a] (Ci per MW(e) y)*	*Radio-nuclide*	*Per unit mass of fuel (Ci t^{-1})*	*Per unit electricity generated[a] (Ci per MW(e) y)*
^{3}H	692	23	^{134}Cs	213 000	7 140
^{85}Kr	11 200	375	^{137}Cs	106 000	3 550
^{89}Sr	96 000	3 220	^{137m}Ba	99 600	3 340
^{90}Sr	76 600	2 570	^{140}Ba	430	14.4
^{90}Y	76 600	2 570	^{140}La	495	16.6
^{91}Y	159 000	5 330	^{141}Ce	56 700	1 900
^{95}Zr	276 000	9 250	^{144}Ce	770 000	25 810
^{95}Nb	518 000	17 360	^{143}Pr	694	23.3
^{95m}Nb	5 860	196	^{144}Pr	770 000	25 810
^{103}Ru	89 100	2 986	^{147}Pm	99 400	3 330
^{106}Ru	410 000	13 740	^{151}Sm	1 150	38.5
^{103m}Rh	89 100	2 990	^{154}Eu	6 820	229
^{106}Rh	410 000	13 740	^{155}Eu	6 370	214
^{110m}Ag	261	8.7	^{160}Tb	300	10.1
^{125}Sb	8 130	272	^{239}Np	17.4	0.6
^{125m}Te	3 280	110	^{238}Pu	2 810	94
^{127m}Te	6 180	207	^{239}Pu	330	11
^{127}Te	6 110	205	^{240}Pu	478	16
^{129m}Te	6 690	224	^{241}Pu	115 000	3 850
^{129}Te	4 290	144	^{241}Am	200	6.7
^{129}I	0.038	0.0013	^{242}Cm	15 000	503
^{131}I	2.17	0.073	^{244}Cm	2 490	84

Source: Reference 107.
[a] Assuming a reactor thermal efficiency of 0.33.

1. Krypton-85

144. The thermal fission yield for ^{85}Kr is 0.285 per cent for ^{235}U and 0.144 per cent for ^{239}Pu (31), corresponding to activities per unit of thermal energy of 0.415 and 0.210 Ci per MW(t) d or per unit electricity generated of 505 and 255 Ci per MW(e) y for ^{235}U and ^{239}Pu, respectively. Assuming that 41 per cent of the fissions are fissions of ^{239}Pu in LWR fuel with a burn-up of 33 000 MW(t) d per tonne (31), the ^{85}Kr activity generated per unit electricity generated is about 400 Ci per MW(e) y. A more detailed calculation estimates the generation per unit mass to be 11 200 Ci/t^{-1} (107), corresponding to about 375 Ci per MW(e) y, assuming a thermal efficiency of 33 per cent. Almost all (>99 per cent) of the ^{85}Kr generated by fission in zircaloy-clad LWR fuels is expected to be released at the reprocessing plant (80). The operating experience shown in table 24 confirms this expectation.

2. Tritium

145. Most of the tritium separated from fuel during dissolution appears in the liquid-waste stream. Some is carried into the dissolver off-gas stream. For the NFS plant in the United States only about 7 per cent of the tritium in the fuel was released through the stack (25). More recent reprocessing-plant designs call for about 75 per cent of the tritium present in fuel to be released to the air and liquid waste discharges to be prevented entirely (80).

146. A portion of the tritium in fuel elements is immobilized as a solid zirconium compound in the zircaloy cladding. The fraction thus retained with the hulls is a function of the linear power rating of the fuel. For present LWR fuel this amounts to about 15 per cent (80). Of an estimated 19 Ci per MW(e) y generation rate, Grathwohl (39) estimates that 1.2 Ci per MW(e) y are released at the reactor (PWR), 16 Ci per MW(e) y are released at the reprocessing plant and 1.8 Ci per MW(e) y are retained with solid wastes. Similar released amounts are expected in reprocessing HWR fuel, while about one half of this amount is expected for AGR fuel and one tenth for HTGR fuel (39). Operating experience (table 24) shows ^{3}H release in liquid effluents to be about 13 Ci per MW(e) y for Windscale and about 6 Ci per MW(e) y for the NFS plant. An additional 1 Ci per MW(e) y was estimated released in airborne effluents from NFS.

3. Carbon-14

147. There are no data available on actual ^{14}C discharges from fuel reprocessing plants. In this Annex it is assumed that all of the ^{14}C present in spent fuel is discharged in the airborne effluents during reprocessing, probably in the form of CO_2.

4. Iodine

148. A tonne of reference LWR fuel (table 25) contains 2.17 Ci of ^{131}I and 0.038 Ci of ^{129}I, corresponding to normalized contents of approximately 0.073 and 0.0013 Ci per MW(e) y for ^{131}I and ^{129}I, respectively. The ^{131}I content depends very much on the cooling time. In current practice, the fuel is still held for at least 150 d before reprocessing, and in many cases the spent fuel is stored for much longer periods.

149. Airborne release of ^{129}I at the Windscale and NFS reprocessing plants has been reported to be (2-4)10^{-6} Ci per MW(e) y (table 24). The normalized liquid release of ^{129}I at NFS, 2.5 10^{-4} Ci per MW(e) y, corresponds to about 25 per cent of the total in the fuel. Release of iodine to the sea from Windscale can be as much as 75 per cent of the throughput (22). The global collective dose estimates for ^{129}I are thus strongly dependent on the liquid releases. The chemical form of the iodine is of importance, particularly for the airborne releases. Haller (41) has reported a wide variation in chemical form of ^{131}I discharged from a fuel reprocesssing plant.

5. Caesium, ruthenium, strontium

150. Other radionuclides released from fuel reprocessing plants include ^{137}Cs, ^{134}Cs, ^{106}Ru and ^{90}Sr. For the airborne effluents, the decontamination factors (ratio of activity processed to activity released) are of the order of 10^{7}-10^{10}, based on the NFS experience (^{90}Sr, 4 10^{7}; ^{106}Ru, 3 10^{8}; ^{137}Cs, 4 10^{9}; ^{134}Cs, 9 10^{9}). In liquid effluents, somewhat greater fractions of the processed amounts of these fission products were released. For release to fresh-water streams at NFS, the decontamination factors were 10^{5}-10^{6}. Liquid wastes from NFS were passed through a series of three small lagoons before discharge to the stream. This system provided a 30-60 d hold-up of the liquids and, in addition, removed considerable amounts of activity by sedimentation: >90 per cent of ^{144}Ce, ^{147}Pm, ^{95}Zr and ^{239}Pu, and 60-75 per cent of ^{134}Cs, ^{137}Cs and ^{60}Co (89). Releases of ^{106}Ru, ^{125}Sb and ^{90}Sr were less affected by the lagoon hold-up.

151. For the Windscale situation, in which release is to the sea, the achievement of decontamination factors of 10^{2}-10^{3} is practised. Ratios of ^{89}Sr, to ^{90}Sr and ^{103}Ru to ^{106}Ru in liquid discharges indicate fuel-cooling times of the order of 1 y (125). After 1969/70, releases of long-lived radionuclides, particularly ^{137}Cs, from Windscale increased, due primarily to leakage from fuel elements in the storage pools.

6. Plutonium, americium, curium

152. Considerable attention has recently been focused on the transuranic elements, particularly plutonium; large amounts would become available as a result of increased reprocessing. It could be used in the operation of breeder reactors or in thermal reactors as recycle fuel. Included in table 25 are the activities of transuranic isotopes in the fuel elements of LWRs.

153. During reactor operation, ^{239}Pu is formed in the uranium fuel by neutron capture by ^{238}U, followed by relatively rapid beta decays of ^{239}U and ^{239}Np. Generally more than half of the ^{239}Pu undergoes fission, thus contributing to the energy produced by the reactor. A small amount is transformed by successive neutron capture to ^{240}Pu, ^{241}Pu, ^{242}Pu and ^{243}Pu. Beta decay of ^{241}Pu and ^{243}Pu produce ^{241}Am and ^{243}Am which, following subsequent neutron capture and beta decay, give rise to ^{242}Cm and ^{244}Cm. These interactions are illustrated in figure I.

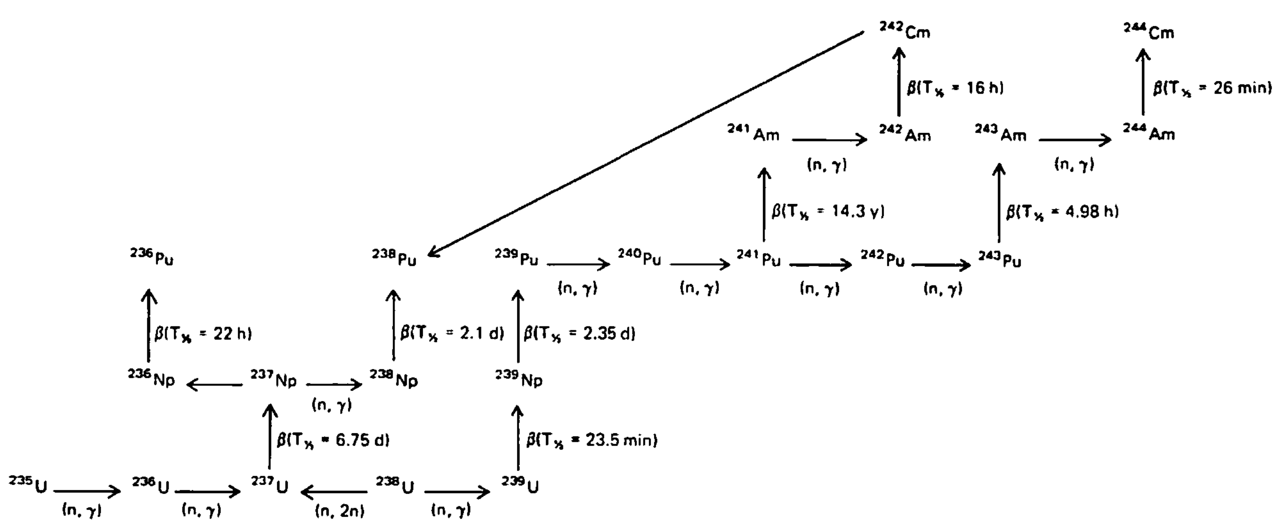

Figure I. Formation scheme for important transuranic radionuclides (31)

154. The isotopic composition of the plutonium in a spent fuel of high burn-up is as follows (weight per cent): ^{239}Pu, 60; ^{240}Pu, 22; ^{241}Pu, 12; ^{242}Pu, 4.5; ^{238}Pu, 1.5 (31). The composition changes with time however. Plutonium-238 activity increases as ^{242}Cm decays, ^{240}Pu increases as ^{244}Cm decays, and ^{239}Pu increases slightly as ^{243}Am decays to ^{239}Np, which subsequently decays to ^{239}Pu. The amounts of ^{241}Am increase significantly as ^{241}Pu decays.

155. Most of the half-lives of the transuranic isotopes are quite long: ^{239}Pu (24 400 y), ^{240}Pu (6580 y), ^{238}Pu (86 y), ^{241}Pu (13.2 y), ^{241}Am (458 y) and ^{244}Cm (17.6 y). Relatively short-lived is ^{242}Cm (163 d), which forms a significant fraction of the alpha activity in spent fuel.

156. Measurements at the NFS fuel reprocessing plant in the United States showed that small amounts of plutonium were discharged in the liquid effluent stream (table 24). In a paper published in 1974, it was suggested that 0.1-1 per cent of the plutonium throughput was being released in liquid effluents from the Windscale reprocessing plant (102).

B. LOCAL AND REGIONAL COLLECTIVE DOSE COMMITMENT

157. The evaluation of the total collective dose from radionuclides released from fuel reprocessing plants requires consideration of the exposure of the local population surrounding the plant and the long-term global exposure. The global dose estimates are presented in a subsequent section. The local environment around the NFS plant, being the first commercial fuel reprocessing plant in the United States, has been studied in considerable detail. A comprehensive review of the population doses from radioactivity discharged from the plant has recently been presented by Martin (93). More detailed surveys to evaluate the doses incurred through the ingestion pathway were made by Magno *et al.* (91). The releases of radioactivity into the Irish Sea from the Windscale plant and the corresponding exposures are still being closely studied (106). Considerable detail in obtaining inventories of activity in the sea and sediments and in tracing the activity in its passage into the North Sea are being achieved (100). Some results of environmental surveys around the reprocessing plant at La Hague have also been published (123).

1. Krypton-85

158. The collective dose commitment from ^{85}Kr released from reprocessing plants is most significant from the global standpoint. As discussed in chapter VI, it is estimated that the collective dose to the gonads due to the release of 375 Ci per MW(e) y of ^{85}Kr is 0.01 man rad per MW(e) y during the first two years, until uniform atmospheric concentrations are achieved. The collective dose received as the ^{85}Kr initially disperses through the local and regional area surrounding the reprocessing plant is much smaller. Using the same local and regional dispersion assumptions as in previous sections of this Annex, namely a dispersion factor of $5\ 10^{-7}$ s m^{-3} at 1 km and a concentration decreasing as the inverse 1.5 power of the distance, a population density of 100 km^{-2}, and a gonadal dose factor (average for testes and ovaries) of $1.1\ 10^{4}$ rad m^{3} Ci^{-1} y^{-1} (104), the gonadal collective dose in the local region (1-100 km from the plant) per unit ^{85}Kr activity released is estimated to be $2\ 10^{-6}$ man rad Ci^{-1}. If all the ^{85}Kr present in the spent fuel were released, the estimated local normalized collective dose to the gonads would be 0.0007 man rad per MW(e) y. For other tissues the dose factors are $1.8\ 10^{4}$, $3.1\ 10^{4}$, $1.8\ 10^{6}$ rad m^{3} Ci^{-1} y^{-1} for red bone marrow, whole lung, and skin, respectively (104). The local normalized collective doses are 0.001, 0.002, and 0.1 man rad per MW(e) y for bone marrow, lung, and skin, respectively.

2. Tritium

159. An estimated 16 Ci of ^{3}H per MW(e) y could be released at fuel reprocessing plants (para. 146), primarily in liquid effluents. For a release of 1 Ci per MW(e) y in airborne effluents, as was the NFS experience (table 24), and assuming a collective dose factor of $2\ 10^{-4}$ man rad per curie released (para. 103), the local collective dose would be $2\ 10^{-4}$ man rad per MW(e) y. For liquid releases the factors for ^{3}H in table 22 can be applied. For the release to salt water (Windscale) of 13 Ci per MW(e) y, the collective dose is $2\ 10^{-6}$ man rad per

MW(e) y, and to fresh water (NFS), of 6 Ci per MW(e) y, the collective dose estimate is 0.04 man rad per MW(e) y.

3. Carbon-14

160. In paragraph 106 the local collective dose commitments per unit activity released following airborne releases of ^{14}C were estimated to be 0.1, 0.12, 0.37 and 0.40 man rad Ci^{-1} for gonads, lung, bone lining cells, and bone marrow, respectively. For an anticipated release of 0.014 Ci per MW(e) y from the reprocessing plant (paras. 73 and 147) the local collective dose commitments are estimated to be as follows (man rad per MW(e) y): gonads, 1 10^{-3}; lungs, 2 10^{-3}; bone lining cells, 5 10^{-3}; bone marrow, 6 10^{-3}.

4. Iodine

161. Members of the public can incur thyroid doses from ^{131}I discharges in airborne effluents from fuel reprocessing plants. A spent-fuel cooling time of 150 d and a fractional release of the order of 10^{-3} for airborne releases of iodine would imply a release of 7 10^{-5} Ci per MW(e) y, of the same order as releases by PWRs and GCRs and much smaller than BWR releases. The NFS reprocessing plant reported ^{131}I releases of 1 10^{-5} Ci per MW(e) y in 1968 and 0.3 10^{-5} Ci per MW(e) y in 1972. From the Windscale reprocessing plant the ^{131}I release was 9 10^{-3} Ci per MW(e) y in 1972 (table 24), but was lower by factors from 20 to 2000 in subsequent years (106). The local collective dose to the thyroid, using a dose factor averaged for the age distribution given in paragraph 117, is estimated to be 22 man rad Ci^{-1} for ^{131}I released in airborne effluents. The local collective dose per unit of electricity generated would therefore be expected to be of the order of 0.002 man rad per MW(e) y for spent fuel held 150 d, and about 100 times more for the highest reported releases in reprocessing plants. Since large amounts of fuel are being held in storage for much longer periods than 150 d, local doses from ^{131}I should generally be insignificant when this fuel is reprocessed.

162. The dosimetry of ^{129}I was considered in table 18 and the thyroid doses based on representative food consumption estimates were given in table 19. The thyroid dose per unit integrated ^{129}I concentration in air through the inhalation and ingestion pathways, weighted for the assumed population age distribution (para. 117), is 1700 mrad per pCi y m^{-3}, which is the same as for the adult thyroid. Using the same dispersion assumptions and population density as in paragraph 103, it is possible to estimate the local collective dose from ^{129}I in airborne effluents to be 300 man rad Ci^{-1}. This is approximately equal to the local collective dose commitment provided that soil accumulation does not greatly increase the levels of ^{129}I in plants. Soldat estimates that soil accumulation of ^{129}I adds 1 per cent per year to the leafy vegetable pathway and 1.3 per cent per year for milk consumption, assuming that all of the iodine in soil is retained in the root zone (132). Kühn *et al.* (79), however, report that iodine penetrates into soil more rapidly and more deeply than other fission products. The global contribution to the collective dose commitment from ^{129}I released to the environment is considered in chapter VI.

163. For an ^{129}I airborne release (table 24) of 4 10^{-6} Ci per MW(e) y, the local collective dose commitment to thyroid is 0.001 man rad per MW(e) y. The local and regional collective dose per unit release of ^{129}I in liquid effluents is 1100 man rad Ci^{-1} for fresh water and 0.7 man rad Ci^{-1} for salt water (table 20). For the reprocessing plant release experience of 2.5 10^{-4} Ci per MW(e) y to fresh water (table 24), and 9.8 10^{-4} Ci per MW(e) y to salt water (para. 149), the local and regional collective dose is estimated to be 0.3 and 0.0007 man rad per MW(e) y for fresh and salt water releases, respectively.

5. Caesium, ruthenium, strontium

164. The fission products ^{137}Cs, ^{134}Cs, ^{106}Ru, ^{89}Sr and ^{90}Sr are important components of fuel reprocessing plant releases and contributors to the collective dose. Caesium-137 and ^{134}Cs may concentrate in fish and result in total body doses to the consumers of the fish. In the past few years, ^{137}Cs has contributed the largest component of the collective dose from radioactive releases from the Windscale reprocessing plant (100, 106). Ruthenium-106 is known to accumulate in the seaweed *Porphyra* growing near the Windscale discharge area in the North Irish Sea, from which the foodstuff laverbread is produced. However, recent changes in the harvesting practice of the seaweed have considerably reduced the ^{106}Ru contribution to the collective dose (100). Caesium, ^{106}Ru, ^{95}Zr-^{95}Nb and ^{144}Ce in sediments cause external exposures. All of these nuclides may contribute small collective doses from inhalation following airborne release from reprocessing plants.

165. Approximate estimates of local collective doses due to the release of airborne particulates can be obtained by using dose factors derived as for reactor releases (table 20). These dose factors account for direct exposure to the radioactive cloud, exposure by the inhalation and ingestion pathways, and by external exposure to deposited radioactivity. Data on typical releases of radioactive particulates in airborne effluents from reprocessing plants are available from the NFS experience (table 24). These average release amounts are used in the example of local collective dose determination given in table 26. The local collective dose commitments are of the order of 10^{-4} man rad per MW(e) y. Ruthenium-106 is an important contributor to the collective dose commitment to the lung via inhalation and to the total body from exposure to contaminated ground. Caesium-137 and ^{134}Cs also contribute to the total body dose through the ingestion pathway and by external exposure. Strontium-90 is responsible for the dose to bone marrow and bone lining cells through the ingestion pathway.

166. Approximate estimates of the collective doses from radionuclides in liquid effluents can be obtained using the dose factors of table 20 and the reported releases from fuel reprocessing plants (table 24). The results are given in table 27. The collective doses from ^{3}H and ^{129}I in liquid effluents, which were considered separately above, are also listed, as are those from ^{95}Zr, ^{95}Nb, ^{144}Ce, and ^{239}Pu. The estimates are based on the operating experience of the Windscale plant for releases to salt water and of the NFS plant for releases to fresh water. For most radionuclides, they are somewhat

TABLE 26. LOCAL COLLECTIVE DOSE COMMITMENTS FROM PARTICULATES IN AIRBORNE EFFLUENTS FROM A FUEL REPROCESSING PLANT[a]

Radio-nuclide	*Normalized activity released (Ci per MW(e) y)*	*Collective dose (man rad per MW(e) y)*			
		Total body	*Lung (whole)*	*Bone marrow*	*Bone lining cells*
^{106}Ru	$4\ 10^{-5}$	$7.6\ 10^{-5}$	$2.2\ 10^{-4}$	$0.04\ 10^{-5}$	
^{137}Cs	$1\ 10^{-6}$	$6.6\ 10^{-5}$	$0.06\ 10^{-6}$	$0.05\ 10^{-6}$	
^{134}Cs	$8\ 10^{-7}$	$4.1\ 10^{-5}$	$0.06\ 10^{-6}$	$0.06\ 10^{-6}$	
^{90}Sr	$6\ 10^{-5}$		$0.02\ 10^{-4}$	$6.1\ 10^{-3}$	$8.4\ 10^{-3}$
^{89}Sr	$3\ 10^{-6}$		$0.03\ 10^{-6}$	$0.02\ 10^{-5}$	
	Total	$2\ 10^{-4}$	$2\ 10^{-4}$	$6\ 10^{-3}$	$8\ 10^{-3}$

[a] Example based on NFS data, 1971-1972 (table 24).

TABLE 27. COLLECTIVE DOSE FROM RADIONUCLIDES IN LIQUID EFFLUENTS FROM FUEL REPROCESSING PLANTS

Radio-nuclide	*Normalized activity released (Ci per MW(e) y)*	*Collective dose (man rad per MW(e) y)*			
		Total body	*Large lower intestine*	*Bone*	*Thyroid*
		Release to salt water			
^{3}H	13	0.000002			
^{137}Cs	14	0.08			
^{95}Zr	10	0.01	0.006		
^{134}Cs	2.3	0.01			
^{144}Ce	5.5	0.005	0.05		
^{106}Ru	12	0.0003	0.05		
^{95}Nb	9.4	0.0001	0.006		
^{103}Ru	0.5	0.000002	0.00006		
^{90}Sr	6.1			0.01	
^{89}Sr	0.4			0.00001	
^{239}Pu	0.6			0.001	
^{129}I	~ 0.001				0.0007
	Total	0.1	0.1	0.01	0.0007
		Release to fresh water			
^{3}H	6	0.04			
^{137}Cs	0.015	0.03			
^{134}Cs	0.0028	0.004			
^{106}Ru	0.075	0.0004	0.02		
^{90}Sr	0.013			0.4	
^{89}Sr	0.0033			0.0003	
^{239}Pu	$7.4\ 10^{-6}$			0.000002	
^{129}I	$2.5\ 10^{-4}$				0.3
	Total	0.07	0.02	0.4	0.3

larger than the previously published estimates (91, 100, 140), primarily because only small, identifiable, local populations had been taken into account in making the earlier estimates. Generalized environmental dispersion has been assumed in the assessments of table 27, however, and differences from more realistic site-specific analyses are expected. For example, the ^{137}Cs releases in liquid effluents from the Windscale plant during 1972/73 have resulted in an estimated collective dose to the entire fish-consuming population in Western Europe (106) of 3000 man rad (half of which is to the United Kingdom population), corresponding to about 1.2 man rad per MW(e) y.

6. Plutonium, americium, curium

167. Both the inhalation and the ingestion pathways to man of the transuranic elements have been investigated, with important data on widespread environmental effects coming from fallout studies (Annex C). Extremely low concentrations were found in foods; this fact, combined with a very low uptake from the gut, indicates that ingestion contributes very little to body content. Inhalation intake could conceivably be of more significance. Annex C discusses the inhalation pathway and the problems of resuspension of plutonium deposited on the soil. Plutonium in airborne effluents has not been reported for fuel reprocessing plants. However, there are data available on total alpha activity released (106).

168. For releases in liquid effluents, no unusual concentrating mechanisms seem to exist, although some high concentration factors have occasionally been reported (154). Plutonium is largely removed to sediments. Hetherington *et al.* (48) have discussed some of the detailed studies of plutonium behaviour in the marine environment following discharge from the Windscale reprocessing plant. Approximate estimates of the local and regional collective dose commitments are included in table 27.

V. RADIOACTIVE WASTE STORAGE AND DISPOSAL

169. Only a small fraction of the radioactivity generated in the nuclear fuel cycle is released to the environment during normal operations. Except for uranium, plutonium, and certain other useful nuclides present in spent fuel, the radioactivity associated with the fuel cycle is considered waste, which must be subjected to suitable treatment and storage or disposal. Storage is taken to mean any arrangement intended to allow the waste to be retrieved if necessary at any future time; the waste may be temporarily inaccessible, but its position and other details have been documented and the means are available to retrieve it. Disposal implies a relinquishing of control over the waste.

A. LOW- AND INTERMEDIATE-LEVEL WASTE

170. The tailings from uranium mills are one example of low-activity solid waste. Common forms of solid waste arising in most types of reactors include spent ion-exchange resins, air and liquid filters, reactor components (control rods, replaced pressure tubes), reactor instruments, protective clothing and miscellaneous paper, plastic and metal wastes. In addition, miscellaneous wastes (e.g., obsolete buildings and equipment) may arise from any installation.

171. Solid wastes can often be reduced in volume, either by compression or incineration. They are then placed in storage or disposed of at surface- or deep-burial sites. Most nuclear sites have some provision for storage of solid wastes. Some reactors provide for storage of solid wastes throughout the reactor lifetime. The final disposition of these wastes will depend on the decision as to the final method of disposal of the reactor itself.

172. It is common practice to dump low-activity solid waste untreated into trenches and cover with earth. At less suitable sites, or if more hazardous materials are to be buried, additional barriers are needed to retard the transfer of activity from the waste to water supplies. These may be provided by constructing concrete wells or pits, or by incorporating the waste in concrete or bitumen.

173. The limitations of shallow burial have been noted. In particular, if significant quantities of long-lived radioisotopes such as ^{90}Sr and ^{137}Cs are included, satisfactory containment is required for a period sufficient for decay. Surface burial is generally considered unsuitable for waste significantly contaminated with transuranium elements, except as temporary storage (108). Since 1970 it has been a requirement in the United States that all solid wastes with significant contamination of transuranium nuclides be segregated and stored retrievably (142).

174. Suitable conditions for a surface-burial site cannot always be found owing to dense population or unfavourable geological or hydrological conditions. The possibility of using abandoned mine workings and natural geological formations as possible sites has therefore been investigated.

175. One of the most thoroughly investigated repositories is the Asse II salt mine in the Federal Republic of Germany. After acquisition of the mine in 1965, studies were made of the stability of the mine room, pillars and safety roofs, the safety of the shaft, and the hydrogeology of the environs. The completion of the studies and a radiological safety assessment enabled the first batch of low-level wastes to be deposited in 1967. The depth of storage of these wastes is about 750 metres. The mine has been used for both low- and intermediate-level radioactive wastes, which generate negligible heat. It is estimated that this one mine will suffice for the low- and intermediate-level wastes generated in the Federal Republic of Germany until 2000 (109).

176. Out of many sites explored in Austria, a gypsum mine and an artificial cavern of 20 000 m^3 were found suitable for permanent storage of low- and intermediate-level radioactive waste, provided that certain additional construction work was undertaken (108).

177. In the south of Spain, several old mines (in particular the Beta mine north of Cordoba) were also suitable for the permanent storage of low- and intermediate-level wastes.

178. The possibility of using the sea to receive and dilute low- and intermediate-level radioactive wastes has been recognized for several decades. The first deliberate introductions of packaged radioactive material into the sea were probably by the United States into the Pacific in 1946. Since about 1949, the United Kingdom and other European countries have made similar disposals in the deep waters of the north-eastern Atlantic Ocean. As a result of the United Nations Conference on the Law of the Sea in 1958, IAEA has produced recommendations that form the basis for deep-sea disposal operations.

B. HIGH-LEVEL WASTE

179. The high-level liquid waste, generated if fuel is reprocessed, contains more than 99 per cent of the fission product and actinide activity in the fuel, except for plutonium and uranium, which are present to the extent of less than 1 per cent. Although some may be disposed of in the liquid state, it appears likely that the rest will be solidified in some manner to render its handling, transport and storage safer and to facilitate its eventual disposal. By the end of the century, the total high-level liquid waste produced throughout the world concomitantly with about 10^7 MW(e) y of nuclear electric power will be equivalent to about 10^4 m^3 of solidified material.

180. The view has been expressed that present storage methods are satisfactory, both on economic and on safety grounds, as an interim measure for the present and near future.

181. Various disposal strategies for these high-level liquid wastes are discussed in detail in several publications (6, 7, 75, 152, 167). Those alternative strategies which have been subjected to the most

intensive scientific and engineering study on the basis of their promise of providing satisfactory isolation of the wastes from man's environment and of being feasible economically include:

> Solidification, followed by eventual disposal into geological formations
>
> Solidification, followed by eventual disposal under or on to the sea bed
>
> Storage, followed by disposal of the liquid wastes directly into geological formations

These three alternatives remain the subject of intensive scientific and engineering study. In addition, some other methods have been examined for their scientific and economic feasibility including disposal of solidified waste in the polar ice caps; extra-terrestrial disposal; and separation, and nuclear transmutation, of the actinide content of the waste.

182. To date, no national authority has reached a final decision on the precise future strategy that it will follow on storage and disposal of high-level waste. Indeed, no major disposal operation involving high-level waste from nuclear power production has yet been undertaken by any country, and in all countries generating this waste it remains in storage.

183. Doses resulting from occupational exposure in waste-storage operations are covered in Annex E. Because high-level wastes from nuclear power production continue to be maintained in storage by national authorities, with the precise method of eventual disposal undecided, the Committee feels that it is unable at this stage to make an adequate assessment of collective dose commitment for the world population from disposal of these wastes. However, the Committee is satisfied that the collective dose contribution from present waste-storage practices is very small compared to the contributions from other parts of the nuclear fuel cycle.

VI. COLLECTIVE DOSE COMMITMENT FROM GLOBALLY DISPERSED RADIONUCLIDES

184. The radionuclides that contribute to the global collective dose are those which are sufficiently long-lived to achieve widespread distribution in the environment. These include ^{3}H, ^{14}C, ^{85}Kr and ^{129}I. Other longer-lived radionuclides, such as ^{137}Cs and ^{239}Pu, probably become less widely dispersed after deposition in soil or sediments following release in a local region. Consideration of the global dispersion and the potential dose contributions has appeared in several reports (5, 11, 22, 74, 88, 92). Estimates of the individual and collective dose commitments from ^{3}H and ^{85}Kr are becoming fairly well established, as experience has led to a good understanding of their environmental behaviour. For radionuclides discharged in liquid effluents, the methods discussed in chapter IV may be applied as a first approximation.

185. The very long-lived nuclides, ^{14}C (5730 y) and ^{129}I (1.6 10^{7} y), pose a special problem. Rather extensive extrapolation into the future is necessary to cover their mean lifetime. In this regard the concept of the incomplete dose commitment, introduced in Annex A, is of special utility for assessing the future maximum *per caput* annual doses resulting from a continuing but finite practice. Thus, if one can foresee that power production by nuclear fission will have a finite duration, which could be of the order of several hundred years, the incomplete dose commitment can be used to estimate the maximum annual *per caput* dose that will be experienced in the future due to that practice. Although the exercise of computing the complete collective doses can be carried out, assuming an ultimate size for the world population, the significance of the results cannot be readily interpreted.

A. TRITIUM

186. The collective dose to the global population from reactor and reprocessing plant releases of tritium can be assessed from the relationships determined for the ^{3}H released into the environment by nuclear explosions. The dose commitment in that case is estimated to be 2 mrad from 2500 MCi released, or 0.8 10^{-12} rad Ci^{-1} for the northern hemisphere (Annex C). This estimate was derived from measurements of the environmental persistence of fallout ^{3}H in surface waters.

187. A rough estimate of the dose commitment for ^{3}H can be obtained by noting that fallout tritium reached levels of around 4000 pCi l^{-1} in surface waters and that subsequent decline of concentration was exponential with a half-time of 3.2 y (12). This corresponds to an integrated concentration of 18 400 pCi y l^{-1}, or about 20 000 pCi y l^{-1} including levels occurring prior to the maximum concentration. Assuming that the total water intake by man in food, air and drinking water is about 3.1 l d^{-1}, and that the tissue dose per unit activity of tritium intake is 84 rad Ci^{-1} (12), a transfer factor of 9.5 10^{4} rad per Ci y l^{-1} is estimated. The dose commitment from the integrated fallout tritium levels is thus estimated to be about 2 mrad, in agreement with the estimate of the previous paragraph.

188. An alternate approach to estimate the dose commitment (11, 22) consists of assuming that all significant discharges of ^{3}H as tritiated water in the near future will occur in the northern hemisphere, and that the tritiated water will be dispersed in the circulating waters of that hemisphere (about 10^{22} g). It is also assumed that the time of exchange with the southern hemisphere and with waters below the thermocline is substantially larger than the half-life of tritium. This approach leads to a lower estimate for the dose commitment, because the higher concentrations of ^{3}H in surface waters before complete mixing occurs are neglected.

189. The collective dose commitment per unit activity of ^{3}H released can be assessed by the procedures outlined in Annex A. In the time scale of interest, the function describing the population growth can be approximated by an exponential with a *fractional*

increase rate a about 2 per cent per year. The present northern hemisphere population N_0 is estimated to be about 3.2 10^9 (100). The collective dose commitment S^c is related in this case to the dose commitment D^c by the equation

$$S^c = \int_0^\infty \bar{D}(t)\,N(t)\,dt = \frac{\lambda_E D^c N_0}{\lambda_E - a}$$

where λ_E is the effective decrease constant of ^{3}H in surface waters. For assessments based on the model presented in paragraphs 186 and 187, λ_E is equal to 1/4.6 y (the effective mean life). With the value of D^c given in paragraph 186, the collective dose commitment is estimated to be about 3 10^{-3} man rad per curie of tritium released. For assessments based on the model outlined in paragraph 188, λ_E is the decay constant of ^{3}H. The collective dose commitment for this model is about 1 10^{-3} man rad Ci^{-1}, probably an underestimate for the reasons given in paragraph 188.

190. Using the value of 3 10^{-3} man rad Ci^{-1} for the estimated reprocessing plant release of 16 Ci per MW(e) y (para. 146), the collective dose commitment is assessed to be about 5 10^{-2} man rad per MW(e) y. Releases of tritium from reactors (tables 11 and 12) make additional contributions of the order of 1 per cent of this value in the case of BWRs and GCRs, 20 per cent in the case of PWRs, and on the average 130 per cent in the case of HWRs.

B. KRYPTON-85

191. Krypton-85 (10.7 y half-time) decays 99.57 per cent of the time by beta-ray emission (mean energy 0.251 MeV). In 0.43 per cent of the disintegrations a lower-energy beta ray (mean energy 0.0475 MeV) and a photon (0.514 MeV) are emitted. The dose to the gonads and bone marrow is primarily due to photons from ^{85}Kr decay in air. When a person is immersed in a large volume of air containing ^{85}Kr, the dose rates per unit concentration of ^{85}Kr in air are 0.62 10^4, 1.6 10^4, and 1.8 10^4 rad y^{-1} per Ci m^{-3} in testes, ovaries, and red bone marrow, respectively (104). Beta decay of ^{85}Kr in air contributes almost 100 per cent of the dose in skin, and beta rays originating in the airways of lung account for 58 per cent of the lung dose. The doses per unit concentration of ^{85}Kr in air are 1.8 10^6 rad m^3 Ci^{-1} y^{-1} for skin and 3.1 10^4 rad m^3 Ci^{-1} y^{-1} for lung (104).

192. The collective dose commitment from ^{85}Kr generated during nuclear power production is due almost entirely to the release from reprocessing plants, the release from reactors being quite small in comparison. As the solubility of krypton in water is negligible, there is no sink for the krypton released into the atmosphere, and rather uniform concentrations are achieved in a period of about two years. In the 1972 report of the Committee, the distribution of krypton was taken to be almost homogeneous over the surface of the globe and throughout the troposphere. This assumption is substantiated by results of measurements of ^{85}Kr in surface air at different latitudes (33, 112).

193. Machta has developed a model which takes account of dispersion in the first 2 y after release (88). The local wind-rose is used for the first 6 h (about 100 km of travel), air trajectory climatology from 6 h to about 3 d, mean low-level airflow from 3 to 30 d, constituting the first hemispheric pass, and finally north-south and vertical mixing until uniform air concentrations are achieved. For the reprocessing plant release of 375 Ci per MW(e) y, the combined collective dose contribution of these steps in the first two years is about 0.01 man rad per MW(e) y, using an average gonadal dose factor of 1.1 10^4 rad m^3 Ci^{-1} y^{-1}. Variations in the local population around the release point can introduce variations of about 10-20 per cent in the estimated collective dose contribution during the first two years following release, but there is little difference (1-2 per cent) in the subsequent collective dose contribution.

194. From the third year after release on, the air concentration is subject only to radioactive decay. The collective dose commitment can be calculated by the procedures given in Annex A, assuming an exponential population growth in the time scale of interest, with a fractional increase rate a of about 2 per cent per year and a population at the beginning of the third year N_0 of about 3.9 10^9. As 1 Ci of ^{85}Kr after uniform mixing in the atmosphere (4 10^{18} m^3) gives an average gonadal dose rate R = 2.8 10^{-15} rad y^{-1}, the collective dose commitment per unit activity released is

$$S^c = \int_0^\infty \bar{D}(t)\,N(t)\,dt = \frac{R_0 N_0}{\lambda - a} = 2.4\ 10^{-4}\ \text{man rad Ci}^{-1}$$

195. For the reprocessing plant release of 375 Ci per MW(e) y, the activity remaining after two years is about 330 Ci per MW(e) y and the resulting collective dose commitment is about 0.08 man rad per MW(e) y. The total collective gonadal dose commitment from ^{85}Kr, including the contribution of the first two years, is therefore estimated to be 0.09 man rad per MW(e) y. The corresponding values for bone marrow and lung are 0.15 and 0.25 man rad per MW(e) y.

C. CARBON-14

196. The dose commitment from ^{14}C releases from nuclear power installations can be assessed by assuming that the environmental distribution and behaviour will follow that of natural ^{14}C. The dose commitment per unit activity released has been estimated by this procedure to be approximately 4.6 10^{-8} rad Ci^{-1} for the whole body, 2 10^{-8} rad Ci^{-1} for gonads and lung, and about 7 10^{-8} rad Ci^{-1} for endosteal cells and bone marrow (Annexes B and C).

197. As this dose is delivered over a very long time, much longer than the time required by the world population to reach an equilibrium value, it can be shown (Annex A) that the collective dose commitment is, very approximately, given by $S^c \simeq D^c N_\infty$, where N_∞ is the assumed upper bound of the population size. Assuming a value of 10^{10} for N_∞, the collective dose commitment per unit activity release is 460 man rad

Ci^{-1} for the whole body, 200 man rad Ci^{-1} for gonads and lung, and 700 man rad Ci^{-1} for endosteal cells and bone marrow.

198. The combined ^{14}C release from light-water reactors and reprocessing plants (about 0.02 Ci per MW(e) y) gives therefore a whole-body collective dose commitment of 9 man rad MW(e) y, while the value for gonads and lung is about 4 man rad per MW(e) y and the value for endosteal cells and bone marrow is about 14 man rad per MW(e) y. As is discussed in Annex A, it would not be realistic to use these collective dose commitments to assess the maximum *per caput* annual dose in the future, because it cannot be assumed that the practice will continue for the long time required to achieve equilibrium. The incomplete collective dose commitment is the proper quantity for this purpose.

199. Assuming that power production by nuclear fission will last for a few hundred years (for example 500 y) it is possible to calculate the incomplete collective dose commitment by an appropriate model for the carbon cycle in the environment. Using the environmental model that incorporates a thick diffusive layer between the surface and deep oceans, discussed in Annex C, the incomplete whole-body collective dose commitment is estimated to be about 2 man rad per MW(e) y. For gonads and lung, the corresponding value is 0.9 man rad per MW(e) y and for endosteal cells and bone marrow 3 man rad per MW(e) y.

D. IODINE-129

200. The extent to which locally released ^{129}I participates in the global iodine cycle is not certain, but because of its very long half-life ($1.6\ 10^7$ y) it may be considered to become widely distributed. The specific-activity approach has been the usual method to assess the thyroid dose commitment from ^{129}I discharges. A specific activity of 1 pCi g^{-1} stable iodine in thyroid tissue results in an absorbed dose rate of $0.9\ 10^{-7}$ rad y^{-1} in the thyroid of a 6-month-old child, $2.1\ 10^{-7}$ rad y^{-1} in a 4-y-old, $3.5\ 10^{-7}$ rad y^{-1} in a 14-year-old, and $7.2\ 10^{-7}$ rad y^{-1} in the adult thyroid. The absorbed dose rate increases with age because the iodine concentration increases, being 80, 180 and 300 ppm at ages 6 months, 4 y and 14 y respectively (119), and 600 ppm in the adult thyroid (68). An average absorbed dose rate, weighted for an assumed population age distribution (para. 117), is $5.5\ 10^{-7}$ rad y^{-1} per pCi g^{-1}.

201. Stable iodine is non-uniformly distributed in nature. The largest fraction, however, is in sea water. Evaporation of sea water contributes most of the exchange between hydrosphere and atmosphere. Atmospheric water is exchanged within about 10 d, and it follows therefore that there is a rapid exchange of iodine. Assuming 10^{22} g to be the mass of circulating water with an iodine concentration of 0.064 ppm (139), there is an initial mixing capacity for ^{129}I of $6.4\ 10^{14}$ g of stable iodine. Over the mean life of ^{129}I, exchange with the deep ocean must also be considered. Uniform mixing is achieved within 100-200 y. The mixing capacity for ^{129}I is thereby increased by a factor of 60. For computing the dose commitment, therefore, the release of 1 Ci of ^{129}I can be considered to result in a specific activity of $2.6\ 10^{-17}$ Ci g^{-1}, giving an average absorbed dose rate to thyroid of $1.4\ 10^{-11}$ rad y^{-1}.

202. Assuming that about half of the ^{129}I present in spent fuel is discharged in liquid releases (para. 149), that 10^{10} is the upper bound of the world population, and that only decay decreases the availability of ^{129}I, the collective dose commitment to the thyroid is estimated to be

$$S^c = \left(6.5\ 10^{-4}\,\frac{\text{Ci}}{\text{MW(e)}\,\text{y}}\right)\left(1.4\ 10^{-11}\,\frac{\text{rad}}{\text{y Ci}}\right)\left(\frac{1.6\ 10^7\ \text{y}}{0.693}\right)$$
$$(10^{10}\ \text{man}) = 2100\ \text{man rad per MW(e) y}$$

As discussed in the case of ^{14}C, a dose commitment delivered over such an enormous amount of time is of doubtful meaning. In particular, it cannot be used for assessing future maximum *per caput* annual doses resulting from a continuing practice. A more useful quantity in this case, the incomplete collective dose commitment, can be calculated with the assumption that power generation by nuclear fission will last for a few hundred years. Using 500 y as a representative value, the incomplete collective dose commitment to the thyroid is estimated to be 0.05 man rad per MW(e) y.

VII. MISCELLANEOUS CONTRIBUTIONS

203. Several operations conducted in support of fuel cycle installations may make additional contributions to the collective dose commitment from nuclear power production. Examples of such operations are the transportation of materials between fuel cycle installations and some research and development activities.

A. TRANSPORTATION

204. Part of the operating requirement of the nuclear fuel cycle is the need to transport radioactive materials: unirradiated fuel from the fabrication plant to the reactor, spent fuel from the reactor to the reprocessing plant, and solid wastes from the reactor and reprocessing plant to waste storage or burial grounds. During this transportation, two groups of people will receive radiation doses: the transport workers and the general public. The contribution made by the exposure of transport workers to the total collective dose resulting from the generation of nuclear power is discussed in Annex E.

205. Nuclear fuel and radioactive waste materials are usually shipped by truck, rail or barge. The transport of radioactive material is subject to regulations, which in virtually all countries are based on the IAEA model regulations (58). The number of shipments and the shipping distances may vary greatly from installation to installation and from country to country. Typical distances and numbers of shipments to support a large reactor have been published in the United Kingdom (169) and the United States (145).

TABLE 28. ESTIMATED COLLECTIVE DOSE TO THE PUBLIC DUE TO SHIPMENTS TO AND FROM REACTORS

(man rad per MW(e) y)

Country	*Reactor type*	*Population group*[a]	*Unirradiated fuel*		*Irradiated fuel*		*Solid waste*		*Total*
United Kingdom	GCR	A	$2 \; 10^{-5}$	(500)	$1 \; 10^{-4}$	(10^{3})			$4.3 \; 10^{-4}$
		B	$1 \; 10^{-5}$	($1 \; 10^{6}$)	$3 \; 10^{-4}$	($3 \; 10^{6}$)			
United States	LWR	A	$4 \; 10^{-7}$	(60)	$1.2 \; 10^{-3}$	(700)	$9.1 \; 10^{-4}$	(600)	$4.4 \; 10^{-3}$
		B	$1.3 \; 10^{-6}$	($3 \; 10^{5}$)	$1.6 \; 10^{-3}$	($7 \; 10^{5}$)	$6.5 \; 10^{-4}$	($3 \; 10^{5}$)	

Note: The number in parentheses is the number of individuals involved.

[a] A = individuals in close proximity; B = individuals along the route.

206. For purposes of estimating the collective dose, the public may be divided into two groups: those individuals along the route who may come into relatively close proximity to the shipment and those at some distance from the shipment as it passes. The collective doses to the first group are estimated by the same procedure as that described in Annex E. The collective doses to the second group are estimated by calculating the dose rate as a function of distance from the package out to about 1 km, multiplying the average dose rate by the population density along the route and integrating the result over the distance travelled. A summary of the calculated doses to the public is given in table 28.

207. The estimated total collective dose to the public from normal transportation associated with reactor operations is in the range of $4 \; 10^{-4}$ to $4 \; 10^{-3}$ man rad per MW(e) y. The risk of additional radiation exposure from accidents is extremely small. The probability of a transportation accident, of which less than 1 per cent involve a severe impact or fire, is about 10^{-6} per vehicle-mile (145). Combining the low probability of accident occurrence with the low probability of radioactivity release in the event of an accident, it has been concluded that collective doses from transport accidents are an insignificant addition to the collective doses from normal transportation (145).

B. NUCLEAR RESEARCH LABORATORIES

208. A portion of the radioactivity released to the environment from nuclear research laboratories may be attributed to support for continued operation or future development of the nuclear fuel cycle. However, other activities at nuclear centres, such as for example radioisotope production and processing, and other types of research, are usually responsible for a large portion of the radioactive releases.

209. Reported estimates of collective dose to the local population during 1973 from 28 research and operational installations in the United States range from 10^{-6} to 200 man rad (160). The highest values are for the largest research and production centres, Savannah River, Argonne, Mound, and Hanford. The total collective dose from all installations (about 450 man rad) divided by the electrical energy generated by nuclear reactors in the country in the same year gives 0.05 man rad per MW(e) y. A much smaller, but indeterminate value represents the portion of this collective dose incurred only in support of nuclear fuel cycle operations.

210. The collective dose to gonads from radionuclides in liquid effluents from the research reactor installations Winfrith and Dounreay in the United Kingdom have been estimated not to exceed 0.1 and 1 man rad, respectively, during 1972-1973 by the shellfish pathway (100). From the Harwell Laboratories the estimated collective dose is 100 man rad by the drinking water pathway from radioactivity released to the River Thames (100). Contributing to this value is the radioactivity, primarily tritium, released from the radioisotope production plant at Amersham (106).

211. Data on radioactive materials released from the nuclear research centres Jülich and Karlsruhe and from research reactor sites in the Federal Republic of Germany have been reported (55). Only for the Karlsruhe laboratories are the releases of any significance. Individuals in the vicinity of the centre may receive an estimated annual dose of 0.3 mrad due primarily to fish consumption (55). The fraction attributable to support of the nuclear fuel cycle cannot be readily estimated.

212. In general, it can be concluded that this component of the collective dose commitment is very small compared to the contribution of other phases of nuclear power production. The collective dose contribution of research and development through occupational exposure is discussed in Annex E.

VIII. SUMMARY OF COLLECTIVE DOSE COMMITMENTS TO THE PUBLIC FROM NUCLEAR POWER PRODUCTION

213. The preceding sections of this Annex contain the results of the Committee's assessments of the collective dose commitments, excluding the contribution of occupational exposure, associated with nuclear power production. These collective dose commitments, per unit electrical energy generated, are summarized in table 29.

TABLE 29. SUMMARY OF NORMALIZED COLLECTIVE DOSE COMMITMENTS TO THE PUBLIC DUE TO NUCLEAR POWER PRODUCTION

(man rad per MW(e) y)

	Gonads	*Whole lung*	*Thyroid*	*Bone marrow*	*Bone lining cells*
Local and regional contribution					
Mining and milling					
U, Ra, Th[a]	0.00001	0.00004	0.00001	0.00002	0.0002
Rn[a]	0.0008	0.1	0.0008	0.0008	0.0008
Fuel fabrication					
U[a]		0.00001		0.000004	0.00002
Reactor operation[b]					
Atmospheric pathways					
Kr, Xe, ^{41}Ar	0.2	0.2	0.2	0.2	0.2
^{3}H	0.004	0.004	0.004	0.004	0.004
^{14}C	0.0006	0.0007	0.0006	0.002	0.002
^{131}I			0.1		
Cs, Sr, Co, Ru	0.006	0.006	0.006	0.006	0.006
Water pathways					
^{3}H	0.03	0.03	0.03	0.03	0.03
Cs, Co, Mn, I	0.01	0.01	0.02	0.01	0.01
Fuel reprocessing					
Atmospheric pathways					
^{85}Kr	0.0007	0.002	0.0007	0.001	0.001
^{3}H	0.0002	0.0002	0.0002	0.0002	0.0002
^{14}C	0.001	0.002	0.001	0.006	0.005
^{131}I, ^{129}I			0.2		
Cs, Ru, Sr	0.0002	0.0004	0.0002	0.006	0.008
Water pathways					
^{3}H	0.04	0.04	0.04	0.04	0.04
^{129}I			0.3		
Cs, Ru, Sr	0.09	0.09	0.09	0.2	0.2
Transportation					
External	0.003	0.003	0.003	0.003	0.003
Global contribution					
Fuel reprocessing and reactor operation					
^{3}H	0.1	0.1	0.1	0.1	0.1
^{85}Kr	0.09	0.25	0.09	0.15	0.15
^{14}C[a]	0.9	0.9	0.9	3.0	3.0
^{129}I[a]			0.5		
Total	1.5	1.7	2.0	3.8	3.8

[a] The values are the incomplete collective dose commitments. The complete dose commitments delivered over thousands to millions of years are as follows (man rad per MW(e) y):

Mining and milling:
U lung, 37; bone marrow, 65; bone lining cells, 280
Rn lung, 100
Fuel fabrication
U lung, 120, bone marrow, 220; bone lining cells, 920
Global
^{14}C gonads and lung, 4; bone marrow and bone lining cells, 14
^{129}I thyroid, 2100

[b] Values are averages based on the reactor-type distribution given in table 7.

For the contributions of the very long-lived nuclides, the values in the table are the incomplete collective dose commitments, the complete commitments being given in the footnotes. The use of these quantities is discussed in Annex A, the incomplete commitments being of particular interest for assessing the future maximum annual *per caput* doses in the case of continued practices. The other required inputs for these assessments are the nuclear installed capacity and the world population size. Considerable caution, however, is required for extrapolation into the future. The activity discharged per unit electricity generated has generally decreased during the past several years because of improvements in technology and increases in regulatory control. Since it is very difficult to predict what the future course of technology and control will be, the extrapolation of the dose commitments to future power production may not necessarily be valid.

214. The data summarized in table 29 indicate that more than 50 per cent of the collective dose commitment to the world population from nuclear power production is due to the global dispersion of the long-lived radionuclides ^{14}C, ^{85}Kr and ^{3}H. In some countries, technologies are under development for restricting the release of these radionuclides and also

^{129}I. While collection and retention of these radionuclides will reduce the projected global collective dose commitments, it can also present additional radiation sources for potential local and regional exposures from processing operations, handling, transportation, storage and disposal of the collected concentrated radionuclides.

215. The collective dose commitment can be expressed as the period of time during which natural radiation would have to be doubled to give an equivalent dose increase. The collective dose commitments to various organs shown in table 29 for the production of 1 MW(e) y range from about 2 to about 4 man rad. For the present installed capacity of 80 GW(e) (table 7), and assuming a utilization factor of 70 per cent, the whole body collective dose commitment per year of electric energy production is about 1.3 10^5 man rad, corresponding to about 3.5 h of natural radiation exposure.

Appendix

COLLECTIVE DOSE COMMITMENT FROM RELEASE OF RADIOACTIVE MATERIALS DUE TO ACCIDENTS

216. Although accidents involving the release of radioactive material or unplanned personal exposure can occur at any stage of the fuel cycle, most attention has been focused on reactor accidents. This is not because reactor accidents are more likely, but because the potential consequences for the general public of some reactor accidents are much greater than from accidents at other stages in the fuel cycle. In view of the low accident frequency and limited consequences of non-reactor accidents, only reactor accidents will be covered in this appendix.

217. So far, in several hundred reactor-years of operation, there has not been an accident in a commercial power reactor which has led to any exposure of the public. The frequency of accidents involving radioactivity in the entire nuclear industry, including research and development, has been very low and most of those which have occurred have been at experimental or pilot plants. Of these, only a few have involved exposures of members of the public.

218. Although no power reactor accidents involving exposure of the public have occurred, the commitment to a large nuclear programme implies a commitment to a certain accident probability. In this sense, therefore, there exists an "accident dose commitment" which is incurred by the operation of power reactors. Since the past experience does not provide estimates of the frequency of extremely unlikely events, any accident analysis must be based on a probability approach which takes into account reliability considerations.

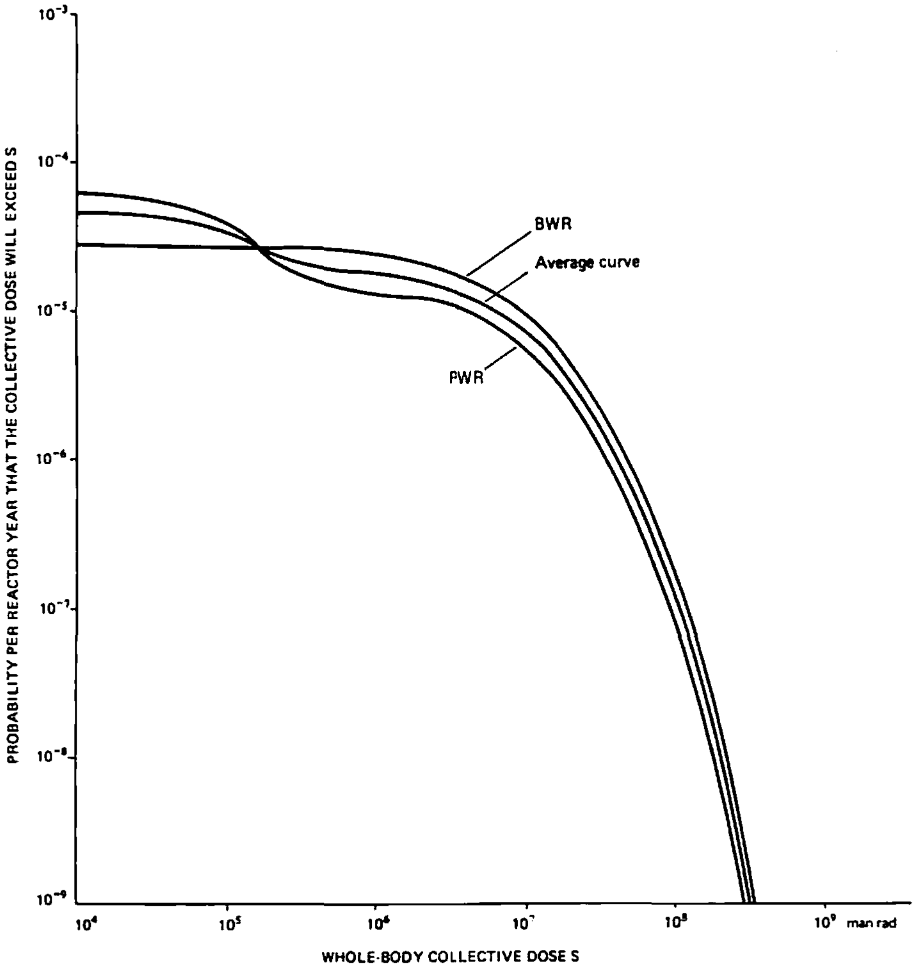

Figure II. Probability distribution for the whole-body collective dose per reactor year. Reactor of 1000 MW(e). (Uncertainties can be accommodated by allowing variations in the average curve of factors of 1/3 and 6 in probability and 1/3 and 3 in consequence.) (155)

219. The most comprehensive analysis to date (155), based on a detailed study of the sequence of events or faults that might lead to release, expressed its main conclusions in terms of the probabilities of different accident releases per reactor-year of operation of large light-water reactors. The doses and collective doses that would result from a given release were assessed by the use of a consequence model, also stochastic in nature, which took account of different weather conditions and population distributions.

220. The probability of incurring an accidental collective dose of a given value (or within a given range) can be obtained from the convolution of the functions representing the release model and the consequence model. An example of such a combined function is represented in figure II, which gives the probability of exceeding selected collective dose values.

221. In the example quoted above, the probability density function (the probability increment per unit collective dose) $f(S)$ can be obtained by differentiation of the function of figure II. The accident collective dose commitment S^c_{acc}, defined as the expectation of collective dose due to accidents, is the integral

$$S^c_{acc} = \int_0^\infty S f(S) dS$$

where, in practice, the integration is carried out over the possible range of collective doses. For large light-water reactors (1000 MW(e)), this integration gives a value of about 250 man rad per reactor-year, implying a normalized accident collective dose commitment of 0.25 man rad per MW(e) y.

222. It should be pointed out that the value quoted above includes individual doses for which proportionality of dose and risk does not apply, e.g., in the case of acute effects. Therefore, it is not possible to assess the relative detriment by comparing this accident collective dose to the collective dose from normal operations.

REFERENCES

1. Aten, J. B. Personal communication (1976).

2. Atkins, D. H. F., R. C. Chadwick and A. C. Chamberlain. Deposition of radioactive methyl iodide vegetation. Health Phys. 13: 91 (1967).

3. Aurand, K., J. Arndt, I. Gans *et al.* Abgaben radioaktiver Stoffe mit dem Abwasser aus Kernkraftwerken in der Bundesrepublik Deutschland. Institut für Wasser-, Boden- und Lufthygiene des Bundesgesundheitsamtes. Bericht WaBoLu 18/74 (1975).

4. Barry, P. J. Estimating dose commitments to populations from radioactive waste disposals into large lakes, p. 499-504 *in* Environmental Behaviour of Radionuclides Released in the Nuclear Industry. IAEA publication STI/PUB/345. Vienna, 1973.

5. Barry, P. J. Energy for tomorrow—nuclear power and the environment. Atomic Energy of Canada, Chalk River Nuclear Laboratories report AECL-5011 (1974).

6. Battelle Pacific Northwest Laboratories. Overview of high level radioactive waste management studies. Report BNWL-1758 (1973).

7. Battelle Pacific Northwest Laboratories. High-level radioactive waste management alternatives. Report BNWL-1900 (1974).

8. Beaujean, H., J. Bohnenstingl, M. Laser *et al.* Gaseous radioactive emissions from reprocessing plants and their possible reduction, p. 63-76 *in* Environmental Behaviour of Radionuclides Released in the Nuclear Industry. IAEA publication STI/PUB/345. Vienna, 1973.

9. Beck, H. L. New perspectives on low level environmental radiation monitoring around nuclear facilities. Nucl. Technol. 14: 323 (1972).

10. Beck, H. L., J. De Campo and C. Gogolak. *In situ* Ge(Li) and NaI(Tl) gamma ray spectrometry. Health and Safety Laboratory report HASL-258 (1972).

11. Beninson, D. Population doses resulting from radionuclides of world-wide distribution, p. 227-231 *in* Population Dose and Standards for Man and His Environment. IAEA publication STI/PUB/375. Vienna, 1974.

12. Bennett, B. G. Environmental tritium and dose to man, p. 1047-1053 *in* Proceedings of the Third International Congress of the International Radiation Protection Association, Washington, 1973. U.S. Atomic Energy Commission report CONF-730907 (1974).

13. Bernhard, S. R. A human metabolic model for ^{14}C–labelled metabolites useful in dose estimation, p. 1400-1405 *in* Proceedings of Third International Congress of the International Radiation Protection Association, Washington, 1973. U.S. Atomic Energy Commission report CONF-730907 (1974).

14. Beverly, R. G. Waste management and milling of uranium. International Symposium on Radiation Protection in Mining and Milling of Uranium and Thorium, Bordeaux, September 1974.

15. Blanchard, R. L. Radiological surveillance studies at the Oyster Creek BWR Nuclear Generating Station. U.S. Environmental Protection Agency report EPA-520/5-76-003 (1976).

16. Bonka, H. and K. Brüsserman. Die Jodfreisetzung aus Kernkraftwerken und Wiederaufarbeitungsanlagen sowie Abschätzung der zukünftigen Umgebungsbelastung. Kernforschungsanlage Jülich report Jül-997-RG (1973).

17. Bonka, H., G. Schwarz and H. B. Wibbe. Contamination of the environment by carbon-14 produced in high temperature reactors. Kerntechnik 15: 297 (1973).

18. Bonka, H., K. Brüssermann and G. Schwarz. Umweltbelastung durch Radiokohlenstoff aus kerntechnischen Anlagen, p. 454-457 *in* Reaktortagung, Deutsches Atomforum, Berlin, 1974.

19. Booth, R. S., S. V. Kaye and P. S. Rohwer. A radiological assessment of radionuclides in liquid effluents of light water nuclear power stations. Oak Ridge National Laboratory report ORNL-TM-4762 (1975).

20. Breslin, A. J. Personal communication.

21. Bryant, P. M., and F. Morley. Disposal of radioactive wastes arising in the United Kingdom from the peaceful uses of atomic energy. National Radiological Protection Board report NRPB-R1, Harwell (1971).

22. Bryant, P. M. and J. A. Jones. The future implications of some long-lived fission product nuclides discharged to the environment in fuel reprocessing wastes, p. 131-153 *in* Management of Radioactive Wastes from Fuel Reprocessing. Proceedings of a symposium jointly organized by OECD/NEA and IAEA. Paris, 1972.

23. Bryant, P. M. and J. A. Jones. Estimation of radiation exposure associated with inert gas radionuclides discharged to the environment by the nuclear industry, p. 91-104 *in* Environmental Behaviour of Radionuclides Released in the Nuclear Industry. IAEA publication STI/PUB/345. Vienna, 1973.

24. Clarke, R. H. An assessment of individual and collective doses due to ^{41}A discharges from CEGB Magnox reactors. Central Electricity Generating Board report RD/B/N 3483 (1975).

25. Cochran, J. A., W. R. Griffin and E. J. Troianello. Observation of airborne tritium waste discharge from a nuclear fuel reprocessing plant. United States Environmental Protection Agency report EPA/ORP 73-1 (1973).

26. Comisión Nacional de Energía Atómica, Argentina. Communication (1976).

27. Commission of the European Communities. Radioactive effluents from nuclear power stations in the community. Discharge data–radiological aspects. Directorate for Health Protection, April 1974.

28. Culot, M. V. J. and K. J. Schiager. Radon control in buildings. Final report of the U.S. Environmental Protection Agency grant No. R 01-EC 00153. Colorado State University, Fort Collins, Colorado, May 1973.

29. Davis, W. Carbon-14 production in nuclear reactors. Oak Ridge National Laboratory report ORNL/NUREG/TM-12 (1977).

30. Denham, D. H. and J. K. Soldat. A study of selected parameters affecting the radiation dose from radionuclides in drinking water downstream of the Hanford project. Health Phys. 28: 139 (1975).

31. Erdman, C. A. and A. B. Reynolds. Radionuclide behavior during normal opeation of liquid metal cooled fast breeder reactors. Part 1: Production. Nucl. Saf. 16: 43 (1975).

32. Eschrich, H., J. van Geel and E. Detilleux. Management of fuel reprocessing wastes at Eurochemie, p. 313-331 *in* Management of Radioactive Wastes from Fuel Reprocessing. Proceedings of a symposium jointly organized by OECD/NEA and IAEA. Paris, 1972.

33. Farges, L., F. Patti, R. Gros *et al.* Activité du krypton-85 dans l'air des hémisphères Nord et Sud. J. Radioanal. Chem. 22: 147-155 (1974).

34. Fluss, M. J., N. D. Dudey and R. L. Malewicki. Tritium and alpha-particle yields in fast and thermal fission of uranium-235. Phys. Rev. C 6 (6): 2252 (1972).

35. Freke, A. M. A model for the approximate calculation of safe rates of discharge of radioactive wastes into marine environments. Health Phys. 13: 743-758 (1967).

36. Gera, F. Geochemical behavior of long-lived radioactive wastes. Oak Ridge National Laboratory report ORNL-TM-4481 (1975).

37. Gogolak, C. V. and K. M. Miller. Determination of gamma-ray exposure in the vicinity of a boiling water reactor, p. 207-214 *in* Population Exposures. Proceedings of the Eighth Midyear Topical Symposium of the Health Physics Society, Knoxville, October 1974. U.S. Atomic Energy Commission report CONF-741018 (1974).

38. Goldfinch, E. P. A review of the first seven years of operational health physics at Dungeness 'A' (Magnox) nuclear power station, p. 364-369 *in* Proceedings of the Third International Congress of the International Radiation Protection Association, Washington, 1973, U.S. Atomic Energy Commission report CONF-730907 (1974).

39. Grathwohl G. Erzeugung und Freisetzung von Tritium durch Reaktoren und Wiederaufarbeitungsanlagen und die voraussichtliche radiologische Belastung bis zum Jahr 2000. Kernforschungszentrum Karlsruhe report KFK-Ext. 4/73-36 (1973).

40. Grimwood, P. D. and G. A. M. Webb. Radiological protection implication of the disposal of solidified high level waste on the floor of the deep ocean. National Radiological Protection Board report NRPB-R48 Harwell (1977).

41. Haller, W. A. and R. W. Perkins. Organic iodine-131 compounds released from a nuclear fuel chemical reprocessing plant. Health Phys. 13: 733-738 (1967).

42. Hanna, G., D. Primeau and Tunnicliffe. Thermal neutron cross sections and resonance integrals of the reaction ^{17}O (n, α)^{14}C, ^{36}Ar (n, α)^{33}S and ^{14}N (n, p)^{14}C. Can. J. Phys. 39: 1784 (1961).

43. Haubelt, R. Zusammenstellung von Daten über die Jahresabgaben radioaktiver Stoffe mit der Abluft von Kernkraftwerken in der Bundesrepublik Deutschland ab der Inbetriebsnahme und in den USA in den Jahren 1967-1972. Bundesgesundheitsamt, Federal Republic of Germany, Institut für Strahlenhygiene report STH-10/74 (1974).

44. Haubelt, R. Zusammenstellung von Daten über die Jahresabgaben radioaktiver Stoffe mit der Abluft von Kernkraftwerken in der Bundesrepublik Deutschland für das Jahr 1974 und in den USA für das Jahr 1973. Bundesgesundheitsamt, Federal Republic of Germany, Institut für Strahlenhygiene report STH-4/75 (1975).

45. Hayes, D. W. and K. W. MacMurdo. Carbon-14 production by the nuclear industry and its possible effects. Savannah River Laboratory report DP-1374 (1974).

46. Health Physics Society. Symposium on Population Exposure. Proceedings of the Eighth Midyear Topical Symposium of the Health Physics Society Knoxville, 1974. U.S. Atomic Energy Commission report CONF-741018 (1974).

47. Heinemann, K., M. Stoeppler, K. J. Vogt and L. Angelletti. Untersuchungen zur Ablagerung und Desorption von Jod auf Vegetation. Kernforschungsanlage Jülich report Jül-1287 (1976).

48. Hetherington, J. A. and D. F. Jefferies. Some investigations into the behavior of plutonium in the marine environment, p. 193-209 *in* Impacts of Nuclear Releases into the Aquatic Environment. IAEA publication STI/PUB/406. Vienna. 1975.

49. Hilbert, F. Erzeugung und Freisetzung von radioaktiven Krypton- und Xenonisotopen durch Kernreaktoren und Wiederaufarbeitungsanlagen und die voraussichtliche radiologische Belastung bis zum Jahr 2000. Kernforschungszentrum Karlsruhe report KFK-2035 (1974).

50. Hill, S. M. B. Environmental monitoring associated with discharges of radioactive waste during 1973 from United Kingdom Atomic Energy Authority establishments. U.K. Atomic Energy Authority report AHSS R3.

51. Hoffman, F. O. Environmental variables involved with the estimation of the amount of iodine-131 in milk and the subsequent dose to the thyroid. Institut für Reaktorsicherheit der Technischen Überwachungs-Vereine e.V. report IRS-W-6, Köln (1973).

52. Hoffman, F. O. A reassessment of the parameters used to predict the environmental transport of iodine-131 from air to milk. Institut für Reaktorsicherheit der Technischen Überwachungs-Vereine e.V. report IRS-W-13 Köln (1975).

53. Hoffman, F. O. A reassessment of the deposition velocity in the prediction of the environmental transport of radioiodine from air to milk. Health Phys. (to be published).

54. Horbert, M., K. J. Vogt and L. Angeletti. Untersuchung zur Ablagerung von Aerosolen auf Vegetation und anderen Grenzflächen. Kernforschungsanlage Jülich report Jül-1288 (1976).

55. Huber, O., I. Gans, R. Wolter and R. Haubelt. Strahlenexposition im Jahre 1974 durch kerntechnische Anlagen in der Bundesrepublik Deutschland. Institut für Strahlenhygiene des Bundesgesundheitsamtes, Bericht STH-5/76 (1976).

56. Hutchinson, J. A., J. C. Daly, M. G. Virgil *et al.* Corrosion products in nuclear reactor effluents. Health Phys. 29: 926 (1975).

57. International Atomic Energy Agency. Directory of Nuclear Reactors. Power Reactors. Vols. IV, VII and IX. IAEA publications STI/PUB/53, 174, 296. Vienna, 1962, 1968, 1971.

58. International Atomic Energy Agency. Safety series no. 6. First published 1961, revised 1964, 1967 and 1973.

59. International Atomic Energy Agency. Radioactive Contamination of the Marine Environment. IAEA publication STI/PUB/313. Vienna, 1973.

60. International Atomic Energy Agency. Nuclear Power and the Environment. IAEA publication STI/PUB/321. Vienna, 1973.

61. International Atomic Energy Agency. Principles and Standards of a Reactor Safety. IAEA publication STI/PUB/342. Vienna, 1973.

62. International Atomic Energy Agency. Environmental Behaviour of Radionuclides Released in the Nuclear Industry. IAEA publication STI/PUB/345. Vienna, 1973.

63. International Atomic Energy Agency. Environmental Surveillance around Nuclear Installations. IAEA publication STI/PUB/353. Vienna, 1974.

64. International Atomic Energy Agency. Physical Behaviour of Radioactive Contaminants in the Atmosphere. IAEA publication STI/PUB/354. Vienna, 1974.

65. International Atomic Energy Agency. Population Dose and Standards for Man and his Environment. IAEA publication STI/PUB/375. Vienna, 1974.

66. International Atomic Energy Agency. Power Reactors in Member States, 1976 edition. IAEA publication STI/PUB/423/2. Vienna, 1976.

67. International Atomic Energy Agency. Impacts of Nuclear Releases into the Aquatic Environment. IAEA publication STI/PUB/405. Vienna, 1975.

68. International Commission on Radiological Protection. Report of the Task Group on Reference Man. A report prepared by a task group of Committee 2. ICRP publication 23. Pergamon Press, Oxford, 1975.

69. International Radiation Protection Association. Proceedings of the Third International Congress of the International Radiation Protection Association. Washington, 1973. U.S. Atomic Energy Commission report CONF-730907 (1974).

70. Kahn, B., R. L. Blanchard, H. L. Krieger *et al.* Radiological surveillance studies at a boilding water nuclear power reactor. U.S. Department of Health, Education and Welfare report BRH/DER 70-1 (1970).

71. Kahn, B., R. L. Blanchard, H. L. Krieger *et al.* Radiological surveillance at a pressurized water nuclear power reactor. U.S. Environmental Protection Agency report RD 71-1 (1971).

72. Kahn, B. *et al.* Radiological surveillance study at the Haddam Neck PWR nuclear power station. U.S. Environmental Protection Agency report EPA-520/3-74-007 (1974).

73. Kautsky, H. Vorkommen und Ausbreitung radioaktiver Stoffe im Meer, p. 432-443 *in* Strahlenschutz und Umweltschutz, Band II, Jahrestagung 1974, Fachverband für Strahlenschutz, 1974.

74. Kelly, G. N., J. A. Jones, P. M. Bryant *et al.* The predicted radiation exposure of the population of the European Community resulting from discharges of krypton-85, tritium, carbon-14 and iodine-129 from the nuclear power industry to the year 2000. Directorate of Health Protection, Commission of European Communities document V/2676/75 (1975).

75. Klett, R. D. Deep rock nuclear waste disposal test: design and operation. Sandia Laboratories Energy report SAND-74-0042 (1974).

76. Kouts, H. and J. Long. Tritium production in nuclear reactors, p. 38-45 *in* Tritium (A. A. Moghissi and M. W. Carter, eds.). Messenger Graphics, Las Vegas, Nevada, 1973.

77. Krause, H. and R. Randl. Treatment and final disposal of radioactive wastes from fuel reprocessing in the Federal Republic of Germany: a survey on policy and R and D work, p. 199-227 *in* Management of Radioactive Wastes from Fuel Reprocessing. Proceedings of a joint OECD/NEA and IAEA symposium, Paris, 1972.

78. Kristan, J., I. Kobal and F. Legat. Measurements of radon under various working conditions in the explorative mining of uranium, p. 217-221 *in* Population Dose and Standards for Man and his Environment. IAEA publication STI/PUB/375. Vienna, 1974.

79. Kühn, W., J. Handl and H. P. Schätzler. Transport of ^{131}I, ^{137}Cs, ^{106}Ru, ^{144}Ce and ^{54}Mn in an undisturbed soil under natural environmental conditions, p. 361-364 *in* Environmental Behaviour of the Radionuclides Released in the Nuclear Industry. IAEA publication STI/PUB/345. Vienna, 1973.

80. Kullen, B. J., L. E. Trevorrow and M. J. Steindler. Tritium and noble gas fission products in the nuclear fuel cycle. II. Fuel reprocessing plants. Argonne National Laboratory report ANL-8135 (1975).

81. Kunz, C., W. E. Mahoney and T. W. Miller. Carbon-14 gaseous effluent from pressurized water reactors, p. 229-234 *in* Population Exposures. Proceedings of the Eighth Topical Symposium of the Health Physics Society, Knoxville, 1974. U.S. Atomic Energy Commission report CONF-741018 (1974).

82. Kunz, C., W. E. Mahoney and T. W. Miller. Carbon-14 gaseous effluents from boiling water reactors. Proceedings of the Meeting of the American Nuclear Society, New Orleans, 1975.

83. Laser, M., H. Beaujean, P. Filss *et al.* Emission of radioactive aerosols from reprocessing plants, p. 99-106 *in* Physical Behaviour of Radioactive Contaminants in the Atmosphere. IAEA publication STI/PUB/355. Vienna, 1974.

84. Lindell, B. Personal communication (1976).

85. Locante, J. and D. D. Malinowski. Tritium in pressurized water reactors, *in* Tritium (A. A. Moghissi and M. W. Carter, eds.). Messenger Graphics, Las Vegas, Nevada, 1973.

86. Lowder, W. M., P. D. Raft and C. V. Gololak. Environmental gamma radiation from nitrogen-16 decay in the turbines of a large boiling water reactor. Health and Safety Laboratory report HASL-271 New York (1973).

87. Lowder, W., P. D. Raft and G. deP. Burke. Determination of ^{16}N gamma radiation fields at BWR nuclear power stations. Health and Safety Laboratory report HASL-305, New York (1976).

88. Machta, L., G. J. Ferber, and J. L. Hefter. Regional and global scale dispersion of krypton-85 for population-dose calculations, p. 411-424 *in* Physical Behaviour of Radioactive Contaminants in the Atmosphere. IAEA publication STI/PUB/355. Vienna, 1974.

89. Magno, P. J., T. Reavey and J. Apidianakis. Liquid waste effluents from a nuclear fuel reprocessing plant. U.S. Department of Health, Education and Welfare report BRH/NERHL 70-2 (1970).

90. Magno, P. J., C. B. Nelson and W. H. Ellett. A consideration of the significance of carbon-14 discharges from the nuclear power industry, p. 1047-1054 *in* Proceedings of the Thirteenth AEC Air Cleaning Conference, San Francisco, 1974. U.S. Atomic Energy Commission report CONF-740807 (1975).

91. Magno, P. J., R. Krankowski, T. Reavey *et al.* Studies of ingestion dose pathways from the nuclear fuel services fuel reprocessing plant. U.S. Environmental Protection Agency report EPA-520/3-74-001 (1974).

92. Martin, A. and H. A. Simon. Population exposure and the interpretation of its significance, p. 15-24 *in* Population Dose and Standards for Man and His Environment. IAEA publication STI/PUB/375. Vienna, 1974.

93. Martin, J. A. Calculations of environmental radiation exposures and population doses due to effluents from a nuclear reprocessing plant. Radiat. Data Rep. 14 (2): 59-76 (1973).

94. Martin, J. A., M. A. Culliton and C. Robbins. Calculations of doses, population doses and potential health effects due to atmospheric releases of radionuclides from U.S. nuclear power reactors in 1972. Radiat. Data Rep. 15 (8): 477-482 (1974).

95. Martin, J. A., C. B. Nelson and H. T. Peterson. Trends in population radiation exposure from operating BWR gaseous effluents, p. 181-190 *in* Population Exposures. Proceedings of the Eighth Midyear Topical Symposium of the Health Physics Society, Knoxville, October 1974. U.S. Atomic Energy Commission report CONF-741018 (1974).

96. Martin, J. A. and C. B. Nelson. Calculations of dose and population dose in general environment due to boiling water nuclear power reactor radionuclide emissions in the United States in 1971, p. 319-333 *in* Physical Behaviour of Radioactive Contaminants in the Atmosphere. IAEA publication STI/PUB/355. Vienna, 1974.

97. Martin, J. J. and B. Chaperon. Rejet d'effluents gazeux et liquides et production de déchets solides des centrales nucléaires d'Electricité de France. VIIe Congrès international de la Société française de radioprotection, Versailles, mai 1974.

98. Miller, K. M., C. V. Gogolak and P. D. Raft. Final report on continuous monitoring with high pressure argon ionization chambers near the Millstone Point boiling water power reactor. Health and Safety Laboratory report HASL-290, New York (1976).

99. Misono, K. Personal communication (1976).

100. Mitchell, N. T. Radioactivity in surface and coastal waters of the British Isles, 1972-1973. U.K. Ministry of Agriculture, Fisheries and Food. Fisheries Radiobiological Laboratory reports FRL-7, FRL-8, FRL-9, FRL-10 (1971, 1973, 1975).

101. Mitchell, N. T. and D. F. Jefferies. Control of low-level liquid radioactive waste disposal: UK experience in the derivation and use of environmental capacity, p. 633-644 *in* Environmental Behaviour of Radionuclides Released in the Nuclear Industry. IAEA publication STI/PUB/345. Vienna, 1973.

102. Morley, F. and G. N. Kelly. Radiological protection and transuranic wastes from the nuclear fuel cycle. Proceedings of a NEA/CEA Seminar on the management of plutonium contaminated solid wastes, Marcoule, October 1974.

103. Nagel, E. Measurement and estimation of gamma-ray doses as a result of emission of argon-41 into the atmosphere, p. 273-275 *in* Physical Behaviour of Radioactive Contaminants in the Atmosphere. IAEA publication STI/PUB/355. Vienna, 1974.

104. National Council on Radiation Protection and Measurements. Krypton-85 in the atmosphere—accumulation, biological significance, and control technology. NCRP report No. 44 (1975).

105. National Institute of Radiation Protection, Sweden. Communication (1976).

106. National Radiological Protection Board. The data submitted by the United Kingdom to the United Nations Scientific Committee on the Effects of Atomic Radiation for the 1977 Report to the General Assembly. NRPB report R47, Harwell (1976).

107. Oak Ridge National Laboratory. Siting of fuel reprocessing plants and waste management facilities. Oak Ridge National Laboratory report ORNL-4451 (1970).

108. Organisation for Economic Co-operation and Development. Radioactive Waste Management Practices in Western Europe. Paris, 1971.

109. Organisation for Economic Co-operation and Development. Management of Radioactive Wastes from Fuel Reprocessing. Proceedings of a symposium, Paris, 1972.

110. Organisation for Economic Co-operation and Development. OECD report referenced in Atomic Energy Rev. 13: 695 (1975).

111. Palms, J. M., V. R. Veluri and F. W. Boone. The environmental impact of iodine-129 released by a nuclear fuel reprocessing plant. Nucl. Saf. 16: 593-602 (1975).

112. Pannetier, R. Distribution transfert atmosphérique et bilan du krypton-85. Commissariat à l'énergie atomique report CEA-R-3591 (1968).

113. Passant, F. H. Environmental radiation from advanced gas cooled reactors in normal operation, p. 326-331 *in* Proceedings of the Third International Congress of the International Radiation Protection Association. Washington, 1973. U.S. Atomic Energy Commission report CONF-730907 (1974).

114. Pelletier, C. A., J. E. Cline and J. H. Keller. Measurements of sources of iodine-131 release to atmosphere from nuclear power plants. IEEE Transactions on Nuclear Science, NS-21: 478 (1974).

115. Pepper, R. B. and A. Short. Discharges of radioactive wastes from CEGB nuclear power stations during 1973, and the results of associated environmental monitoring. Central Electricity Generating Board report NHS/R-122/74 (1974).

116. Perkins, R. W. Physical and chemical form of iodine-131 in fallout. Health Phys. 9: 113 (1963).

117. Poole, L. G. World uranium resources. Nuclear Eng. Int. 20(224): 95-100 (1975).

118. Porz, F. Erzeugung und Freisetzung von radioaktiven Jodisotopen durch Kernreaktoren und Wiederaufarbeitungsanlagen und die voraussichtliche radiologische Belastung bis zum Jahr 2000. Kernforschungszentrum Karlsruhe report KFK-1912 (1974).

119. Rohwer, P. S., K. E. Cowser, W. S. Snyder *et al.* An approach for assessing potential population exposures, *in* Environmental Surveillance in the Vicinity of Nuclear Facilities (E. C. Reining, ed.). C. C. Thomas, Springfield, Illinois, 1970.

120. Rothschild, H. C. Personal communication.

121. Rublevsky, V. P., A. S. Zykova and A. D. Turkin. Atomic electric power stations as sources of carbon-14 discharge, p. 296-300 *in* Proceedings of the Third International Congress of the International Radiological Protection Association, Washington, 1973. U.S. Atomic Energy report CONF-730907 (1974).

122. Ruf, M. Zur Radioökologie der Donau, p. 403-431 *in* Strahlenschutz und Umweltschutz, Band II. Fachverband für Strahlenschutz, 1974.

123. Scheidhauer, J., R. Ausset, J. Planet *et al.* Programme de surveillance de l'environnement marin du centre de La Hague, p. 347-365 *in* Population Dose Evaluation and Standards for Man and His Environment IAEA publication STI/PUB/375. Vienna, 1974.

124. Shiager, K. J. Analysis of radiation exposures on or near uranium mill tailings piles, Rad. Data Rep. 11: 411-425 (1974).

125. Schiesser, D. Erzeugung und Freisetzung von ^{89}Sr, ^{90}Sr, ^{103}Ru, ^{106}Ru, ^{134}Cs, ^{135}Cs, ^{137}Cs, ^{141}Ce, und ^{144}Ce durch Kernreaktoren und Wiederaufarbeitungsanlagen durch die voraussichtliche radiologische Belastung bis zum Jahr 2000. Kernforschungszentrum Karlsruhe report KFK-2153 (1975).

126. Sears, M. B., R. E. Blanco, R. C. Dahlman *et al.* Correlation of radioactive waste treatment costs and the environmental impact of waste effluents in the nuclear fuel cycle for use in establishing 'As low as practicable' guides—milling of uranium ores. Oak Ridge National Laboratory report ORNL--TM-4903/1 (1975).

127. Shearer, S. D. and C. W. Sill. Evaluation of radon-222 near uranium tailings piles. Health Phys. 17: 77-88 (1969).

128. Smith, J. M. and R. S. Gilbert. Tritium experience in boiling water reactors, p. 57-68 *in* Tritium (A. A. Moghissi and M. W. Carter, eds.), Messenger Graphics, Las Vegas, Nevada. 1973.

129. Snelling, R. N. and S. D. Shearer. Environmental survey of uranium mill tailings pile, Tuba City, Arizona. Radiol. Health Data 10: 475-487 (1969).

130. Snelling, R. N. Environmental survey of uranium mill tailings pile, Monument Valley, Arizona. Radiol. Health Data 11: 511-517 (1970).

131. Snelling, R. N. Environmental survey of uranium mill tailings pile, Mexican Hat, Utah. Radiol. Health Data 12: 17-28 (1971).

132. Soldat, J. K. Radiation doses from iodine-129 in the environment. Health Phys. 30: 61 (1976).

133. Swift, J. J., J. M. Hardin and H. W. Calley. Potential radiological impact of airborne releases and direct gamma radiation to individuals living near inactive uranium mill tailings piles. U.S. Environmental Protection Agency report EPA-520/1-76-001 (1976).

134. Terpilak, M. S. and B. L. Jorgensen. Environmental radiation effects of nuclear facilities in New York State. Nucl. Saf. 16: 204 (1975).

135. Thompson, S. E., C. A. Burton, D. J. Quinn *et al.* Concentration factors of chemical elements in edible aquatic organisms. Lawrence Livermore Laboratory report UCRL-50564/Rev.1 (1972).

136. Trovorrow, L. E., B. J. Kullen, R. L. Jarry *et al.* Tritium and noble gas fission products in the nuclear fuel cycle. I. Reactors. Argonne National Laboratories report ANL-8102 (1974).

137. Tsivoglou, E. C. Environmental monitoring in the vicinity of uranium mills, p. 231-251 *in* Radiological Health and Safety in Mining and Milling of Nuclear Materials. IAEA publication STI/PUB/78. Vienna, 1963.

138. Tsivoglou, E. C., S. D. Shearer, J. D. Jones *et al.* Estimating human radiation exposure on the Animas river. J. Am. Water Works Assoc. 52: 1270-1290 (1960).

139. Turekian, K. K. The oceans, streams and atmosphere, *in* Handbook of Geochemistry (K. H. Wedepohl, ed.). Springer, Berlin, Heidelberg, New York, 1969.

140. United Nations. Ionizing Radiation: levels and effects. A report of the United Nations Scientific Committee on the Effects of Atomic Radiation to the General Assembly, with annexes. United Nations publications, Sales No. E.72.IX.17 and 18, New York, 1972.

141. United States Atomic Energy Commission. Report on releases of radioactivity from power reactors in effluents during 1970. Division of Compliance, 1971.

142. United States Atomic Energy Commission. Plan for the management of AEC-generated radioactive wastes. U.S. Aomic Energy Commission report WASH-1202 (1972).

143. United States Atomic Energy Commission. Report on releases of radioactivity in effluents from nuclear power plants for 1971. Directorate of Regulatory Operations, 1972.

144. United States Atomic Energy Commission. Operating history, U.S. nuclear power reactors, 1971. U.S. Atomic Energy Commission report WASH-1203-72 (1972).

145. United States Atomic Energy Commission. Environmental survey of transportation of radioactive materials to and from nuclear power plants. U.S. Atomic Energy Commission report WASH-1238 (1972).

146. United States Atomic Energy Commission. Report on releases of radioactivity and solid waste from nuclear power plants for 1972. Directorate of Regulatory Operations, 1973.

147. United States Atomic Energy Commission. The safety of nuclear power reactors and related facilities. U.S. Atomic Energy Commission report WASH-1250 (1973).

148. United States Atomic Energy Commission. Final environmental statement ALAP-LWR effluents. U.S. Atomic Energy Commission report WASH-1258 (1973).

149. United States Atomic Energy Commission. Tritium (A. A. Moghissi and M. W. Carter, eds.) Proceedings of a symposium, Las Vegas, 1971. U.S. Atomic Energy Commission report CONF-710809 (1973).

150. United States Atomic Energy Commission. Noble Gases (R. E. Stanley and A. A. Moghissi, eds.). Proceedings of a symposium, Las Vegas, 1973. U.S. Atomic Energy Commission report CONF-730915 (1975).

151. United States Atomic Energy Commission. Environmental survey of the uranium fuel cycle. U.S. Atomic Energy Commission report WASH-1248 (1974).

152. United States Atomic Energy Commission. High-level radioactive waste management alternatives. U.S. Atomic Energy Commission report WASH-1297 (1974).

153. United States Atomic Energy Commission. Environmental statement LMFBR program (1974).

154. United States Atomic Energy Commission. Plutonium and other transuranium elements: sources, environmental distribution and biomedical effects. U.S. Atomic Energy Commission report WASH-1359 (1974).

155. United States Atomic Energy Commission. Reactor Safety Study. An assessment of accident risks in U.S. commercial power plants. U.S. Atomic Energy Commission report WASH-1400 (NUREG 75/014) (1975).

156. United States Environmental Protection Agency. Environmental analysis of the uranium fuel cycle. Part I. Fuel Supply. U.S. Environmental Protection Agency report EPA-520/9-73-003-B (1973).

157. United States Environmental Protection Agency. Environmental analysis of the uranium fuel cycle. Part II. Nuclear power reactors. U.S. Environmental Protection Agency report EPA-520/9-73-003-C (1973).

158. United States Environmental Protection Agency. Environmental analysis of the uranium fuel cycle. Part III. Nuclear fuel reprocessing. U.S. Environmental Protection Agency report EPA-520/9-73-003-C (1973).

159. United States Environmental Protection Agency. A computer code (RVRDOS) to calculate population doses from radioactive liquid effluents and an application to nuclear power reactors on the Mississippi River basin. U.S. Environmental Protection Agency report ORP/EAD 76-4 (1976).

160. United States Environmental Protection Agency. Radiological quality of the environment. U.S. Environmental Protection Agency report EPA-520/1-76-010 (1976).

161. United States Nuclear Regulatory Commission. Summary of radioactivity released in effluents from nuclear power plants during 1973. U.S. Nuclear Regulatory Commission report NUREG-75/001 (1975).

162. United States Nuclear Regulatory Commission. Final generic environmental statement on the use of recycle plutonium in mixed oxide fuel in light water cooled reactors. U.S. Nuclear Regulatory Commission report NUREG-0002 (1976).

163. United States Nuclear Regulatory Commission. Radioactive materials released from nuclear power plants during 1974. U.S. Nuclear Regulatory Commission report NUREG-0077 (1976).

164. United States Nuclear Regulatory Commission. Methods for estimating atmospheric transport and dispersion of gaseous effluents in routine releases from light-water-cooled reactors. Regulatory Guide I/III (1976).

165. United States Nuclear Regulatory Commission. Calculation of releases of radioactive materials in gaseous and liquid effluents from boiling water reactors (BWR gale code). U.S. Nuclear Regulatory Commission report NUREG-0016 (1976).

166. United States Nuclear Regulatory Commission. Calculations of releases of radioactive materials in gaseous and liquid effluents from pressurized water reactors (PWR gale code). U.S. Nuclear Regulatory Commission report NUREG-0017 (1976).

167. United States Energy Research and Development Administration. Alternatives for managing wastes from reactors and post-fission operations in the LWR fuel cycle. U.S. Energy Research and Development Administration report 76-43 (1976).

168. Vennart, J. and P. J. Ash. Derived limits for ${}^{35}S$ in food and air. Health Phys. 30: 291-294 (1976).

169. Webb, G. A. M. Personal communication (1976).

170. Weiss, B. H., P. G. Voilleque, J. H. Keller. Detailed measurements of iodine-131 in air, vegetation and milk around three operating reactor sites. *in* Environmental Surveillance around Nuclear Installations. IAEA publication STI/PUB/353. Vienna, 1975.

171. Wilson, R. Personal communication (1975).

172. Woodhead, D. S. Level of radioactivity in the marine environment and the dose commitment to marine organisms, p. 499-523 *in* Radioactive Contamination of the Marine Environment. IAEA publication STI/PUB/313. Vienna, 1973.

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