

3. FUEL CYCLE FACILITIES AND THEIR RADIOACTIVE DISCHARGES

49. Wherever possible, data from facilities in Western Europe were taken as these data are generally comprehensive and of direct relevance to the scope of the study. The exception was uranium mining and milling for which there was not an appropriate site in Western Europe

3.1 Mining and Milling

Uranium and mining

50. Uranium is the fuel used in nearly all-existing nuclear reactors. It is present in the earth's crust at low average concentrations of about 1 ppm (part per million by weight or, approximately 0.02 Bq/g soil), but may be concentrated in mineral deposits up to a few percent.

51. Uranium may occur as the predominant metallic constituent (monometallic deposits) or may be accompanied by other metals, particularly Ni, Co and As (polymetallic deposits). As a primary product, uranium bearing ores are generally exploitable at concentrations in excess of several hundred parts per million while as a by-product, uranium can be economically extracted at concentrations around one hundred parts per million or less.

52. Mining of uranium ores is commonly carried out by either underground or open pit techniques. Compared to underground mining, the amount of waste material is larger for open pit methods due to the relatively large volumes involved. A third mining method, in-situ leaching finds more limited application (13 % of worldwide uranium production), owing to the specific prerequisites for this type of process.

53. The uranium is extracted from the crushed ore in a processing plant (mill) using chemical methods appropriate to the specific mineral form. The concentrated form of uranium produced in the plant is called yellow cake (U₃O₈), and contains approx. 70 - 75 % uranium by weight. Depending on its quality, the concentrate is sometimes further purified in a refinery near the mine before being shipped to a conversion plant.

54. The radioactivity of the separated uranium is very low. The radioactive daughter products, with ²³⁰Th as the dominant long-lived radionuclide (half-life of 80 000 years), are left with the mill tailings. They may also contain stable toxic elements such as arsenic, nickel etc. After stabilisation, the tailings are stored adjacent to the mining area on ground or deposited into lakes, mined out pits or ring-dyke impoundments.

Releases from specific facilities

55. During mining and milling, particles and radioactive gases are continuously released to air and to water. Ventilation to reduce exposures from inhaled radionuclides and radiological protection measures against external irradiation are critical parts of any mining operation, in particular with underground mines. The run-off water of mills may contain radionuclides and need treatment before it is released. Seepage from the tailings and waste rock may transport dissolved radionuclides into the ground water. Additionally, the surface water may become contaminated and thereby give result to airborne dispersion.

56. The stored tailings may constitute a source of radon for a long time period. Radon emanation rates from tailings vary considerably depending on the ore grade of the tailings source material, the characteristics of the storage facility and the rehabilitation program.

57. The releases from the mining and milling processes do not occur from a well-defined and monitored point source, thus making the definition of a source term complicated. The mining and milling operations are spread over large areas and the dusts and the gases are released from stacks used to vent the underground tunnels. Particles released from more "closed" facilities may be controlled through filtration but that is not generally applicable to open pit mining.

58. In the UNSCEAR Report 1993, the radon exhalation rate from bare tailings is assumed to range between 10 and 300 Bq per m² and s. After the active surveillance period the applied remedial actions are assumed to restrict the average radon release rate to the level of 3 Bq per m² and s.

59. In a recent study by SENES Consultants (Ref.), information was gathered from eight major uranium production facilities on future releases of radon from their tailings after remediation. In the report, the mean radon release rate for those sites was estimated to range from 0 to 7 Bq m⁻² s⁻¹.

60. Other radionuclides in the decay series that are of potential importance in view of abandoned mill tailings, are ²¹⁰Pb and ²¹⁰Po. These radionuclides may accumulate in the vicinity of the mine and through the food chain cause radiation doses to man. Possible leaching of ²²⁶Ra into the ground water is another example of a potential critical pathway, giving rise to exposures in the far future.

61. The environment around the mines usually exhibits high natural background radiation due to the uranium in the ore, which makes it difficult to separate the releases and the exposures, caused by the mining, from the background. Typical natural background concentrations in the area around sites in Canada range from 0,001 Bq/g to 0,05 Bq/g.

Reference mining and milling facility and generic discharges

62. Data for the generic study were taken from four modern uranium mining and milling facilities, Key Lake and Cluff Lake in Canada, and Ranger and Olympic Dam in Australia. On the basis of these data it was assumed that the uranium tailings occupied an area of 100 hectares and to release radon²²² at an average rate of 3 Bq m⁻² s⁻¹ during the operational phase. The sensitivity of the results to the assumed radon-222 release rate is discussed in Chapter 6 [*check*]. Tailings may also release radionuclides by leaching into water bodies. Typical concentrations of radionuclides in water bodies close to uranium mining and milling facilities are given in Table 2. They are based on Canadian data

(Ref.). Releases of radon-222 were normalised to GWa of electricity on the basis that the corresponding tailings cover 1 hectare (UNSCEAR 1993).

Table 2: Typical activity concentrations in freshwater bodies close to uranium mining facilities

Radionuclide	Freshwater activity concentration (Bq l ⁻¹)
Ra-226	0.05
Pb-210	0.05
Po-210	0.05

3.2 Uranium Conversion, Enrichment and Fuel Fabrication

Uranium ore concentrate to Uranium hexafluoride conversion

Short description of the process

63. LWRs require uranium enriched with the fissile uranium isotope 235 as nuclear fuel. The enrichment process currently used at an industrial scale requires uranium hexafluoride as feed material. The uranium ore concentrate (UOC) coming from mining and milling therefore has to be converted to uranium hexafluoride, which is solid under the conditions of storage and transportation but volatile during the enrichment steps. There are few facilities for this conversion in OECD countries, the only ones in Europe being in France (COMURHEX Malvesi, Pierrelatte) and in the United Kingdom (BNFL Springfields).

64. Processing of UOC to pure aqueous uranyl nitrate is carried out as a continuous operation. Drums of UOC are tipped into dissolver vessels where the UOC is mixed with nitric acid and water. This stage results in a slurry of uranyl nitrate, with insoluble impurities in suspension, which is fed to a filtration system.

65. This filtration stage results in the production of crude uranyl nitrate, together with a filter cake (solid) consisting of the insoluble impurities and trace amounts of uranium which is sent for disposal in a controlled, authorised fashion.

66. The crude uranyl nitrate product then undergoes a solvent extraction process for purification. This stage generates a raffinate effluent which, if meeting sentencing criteria, is discharged to the trade effluent system and thence, after neutralisation, via the factory outfall to the tidal waters of a river estuary. The majority of radioactivity discharged from the site arises from this process stage.

67. The pure aqueous uranyl nitrate solution is then concentrated and undergoes a denitration process during which it thermally decomposes to UO₃ powder. The UO₃ powder generated is then hydrated, reduced with hydrogen gas to UO₂, and hydrofluorinated with anhydrous hydrogen fluoride vapour to produce UF₄ powder.

68. This powder is then transported by pneumatic pipeline to the UF₆ production facility where it is reacted with fluorine gas (electrolytically generated on site) in a fluid-bed reactor. The gaseous UF₆ is cooled and condenses into a solid, collecting in condenser vessels. Finally it is heated under pressure to liquefy it for run off into transport cylinders, and stored for delivery to customers.

Discharges

69. Experience on discharges is available from both the French and British conversion facilities (see Annex A). For Springfields, the available data cover also the short-lived daughter isotopes in addition to the uranium alpha activity. Therefore these data are taken as reference source term for the purpose of this study.

70. The majority of liquid radioactivity (in Bq) discharged from the Springfields site originates from the conversion of UOC to UF₆, so, it is a maximising assumption to take the total site discharge data as representing the UOC to UF₆ process. For aerial releases, the plants in question do not make up a large percentage of the Springfields site releases: data from these plants show that they discharged (principally as natural uranium alpha activity) approximately 0.7GBq in 1995, 0.7 GBq in 1996 and 0.3GBq in 1997. The releases from Malvesi and Pierrelatte are similar (see Annex A).

Uranium enrichment

Short description of the process

71. Two main routes for uranium enrichment, both processing uranium hexafluoride, have been in wide-spread use for many years: the gaseous diffusion process and the enrichment by gas centrifuges. Both technologies are proven technically sound and safe steps within the nuclear fuel cycle. Operational experience has been good with low occupational doses and very small discharges of radioactive substances to the environment. The main hazard from enrichment facilities, especially for the workers, is the accidental release of chemically toxic uranium hexafluoride. Therefore, the prevention of uranium hexafluoride leaks and the protection of the workforce from toxic effects of uranium hexafluoride and its decomposition products is of paramount importance.

72. The EURODIF gaseous diffusion enrichment plant at Tricastin, France is taken as the reference facility for this study because from this large facility real industrial experience is available for many years. As reference plant for the- centrifuge enrichment technique, the URENCO enrichment facility at Gronau, Germany, has been selected.

73. The depleted Uranium separated at enrichment process is assumed to be temporarily stored waiting for future use in reactors.

Discharges

74. Very small quantities of uranium are vented from the process and auxiliary systems of gaseous diffusion plants to atmosphere. Atmospheric releases from EURODIF in 1996 were 4.5 kg uranium with a total alpha activity of 0.18 GBq [*Ted get 4.5 Kg U238 metal = 0.08 GBq*]. Discharges in liquid form occur from process cleanup operations and auxiliary facilities. Generally these discharges are

similarly low as aerial discharges (see Annex A). Liquid releases from EURODIF in 1996 were only 0.4 kg of uranium. Data from diffusion plants in the United States are slightly higher (few kg. of uranium per year).

75. Radioactive discharges from centrifuge enrichment facilities are very small and even lower as from gaseous diffusion facilities. An example is given for the URENCO enrichment plant at Gronau, Germany in the Annex A.

Table 3: Typical releases from Uranium conversion and enrichment processes (Springfields and EURODIF plants)

Facility	Nuclide	Discharges, GBq y ⁻¹	
		Aerial	Liquid
Conversion	Th-230	ND	5.2 · 10 ¹
	Th-232	ND	1.4
	Th-234	ND	7.2 · 10 ⁴
	Pa-234m	ND	7.2 · 10 ⁴
	U-234	5.7 · 10 ⁻¹	5.7 · 10 ⁻¹
Enrichment	U-234	1.8 · 10 ⁻¹	1.8 · 10 ⁻²

Table 4: Typical combined releases for conversion and enrichment normalised per 1 GWa

Nuclide	Discharges GBq/GWa			
	Aerial once-through	1 recycle	Liquid once-through	1 recycle
Th-230	ND	ND	1.34	1.05
Th-232	ND	ND	3.6 · 10 ⁻²	2.8 · 10 ⁻²
Th-234	ND	ND	1.85 · 10 ³	1.46 · 10 ³
Pa-234m	ND	ND	1.85 · 10 ³	1.46 · 10 ³
U-234	1.9 · 10 ⁻²	1.5 · 10 ⁻²	1.47	1.15

By normalisation to plant throughput and taking into account for the reduction of uranium demand per GWa for the recycling strategy (see flow chart of Figure 1) the releases for conversion and enrichment and for the two considered options have been calculated (annual throughput of Springfields 7000 t uranium oxide, 8500 t uranium at EURODIF).

UO₂ fuel fabrication

76. There are a number of PWR fuel fabrication plants in the OECD areas. Romans plant of Franco-Belge de Fabrication de Combustible (FBFC) is selected as a reference facility for fuel fabrication. 650 tHM was processed in 1997 and this is equivalent to 154.5TWh with a mean burn-up of 30 GWd/t.

77. Romans plant uses a dry process for UF₆ conversion: UF₆ is fed co-currently with water vapor in a furnace where it is vaporised. Hydrogen is sent back-stream and the gas-gas reaction produces powder UO₂. This process has an advantage in reducing liquid and gaseous release into the environment.

78. The UO₂ powder is press-compacted into small cylindrical pellets, then sintered and ground to their final configuration. As a result of this process, the fissile material acquires the physical shape and chemical property suitable for being used as nuclear fuel. The pellets are loaded into zircaloy tubes then both ends are sealed (fuel rod). 264 fuel rods are positioned within a fuel assembly skeleton and a top nozzle is fitted. After inspection, the complete fuel assembly is ready for shipment to power plants.

Discharges/Releases

79. For this study, normalised 1997 data per TWh are used for liquid and gaseous releases. Gaseous and liquid releases from the Romans plant in 1997 are given on Tables 5 and 6 respectively.

Table 5: Gaseous releases from ROMANS, 1997

Radionuclide	Activity released (MBq)	Activity released (MBq/TWh)
U-234	1.34 E+01	8.66 E-02
U-235	5.23 E-01	3.38 E-03
U-238	2.25 E+00	1.45 E-02

Note: Radon results indirectly from the natural decay of uranium. The process of uranium ore conversion to UO₂, however, removes all uranium decay products, including radium, the direct parent of radon. Thus, because radium is removed, and because the very long radioactive decay periods of uranium and several of its daughter products, which precede radon in the decay chain, there is no radium present in fuel, and thus no radon is emitted.

Table 6: Liquid releases from ROMANS, 1997

Radionuclide	Activity released (MBq)	Activity released (MBq/TWh)
U-234	2.17 E +03	1.40 E+01
U-235	8.44 E+01	5.47 E-01
U-238	3.62 E+02	2.34 E+00

MOX fuel fabrication

80. There are three large-scale MOX fuel fabrication plants in operation in the OECD area. MELOX plant is selected for a reference plant. It started operation in 1995 and fabricated 100.3 tHM in 1997, which is equivalent to 23.8 TWh with a mean burn-up of 30 GWd/t. It will further increase its output to 210 tHM/y by year 2000.

81. PuO₂, depleted UO₂ and recyclable scraps in the form of (U-Pu) O₂ are blended in order to obtain the required Pu content. The blended powder is ground and further homogenised, then forgoes the same process as the UO₂ fuel fabrication. In case of MELOX plant, PuO₂ powder is supplied by the reprocessing plant and depleted UO₂ powder by uranium enrichment plant, thus liquid and gaseous releases are reduced.

Discharges/Releases

82. For this study, normalised 1997 data per TWh is used for liquid and gaseous releases. Gaseous and liquid releases from MELOX plant are given on Table 7 and 8.

Table 7: Gaseous releases from MELOX, 1997

Radionuclide	Activity released (MBq)	Activity released (MBq/TWh)
Pu-238	< 6.69 E-02	< 2.81 E-03
Pu-239/240	< 5.84 E-02	< 2.45 E-03

Note: Gaseous effluents are only monitored for Plutonium isotopes.

Table 8: Liquid releases from MELOX, 1997

Radionuclide	Activity released (MBq)	Activity released (MBq/TWh)
Pu 238	< 1.98 E+00	< 8.32 E-02
Pu 239	< 3.36 E-01	< 1.41 E-02
Pu 240	< 4.81 E-01	< 2.02 E-02
Pu 241	< 1.67 E-03	< 7.02 E-05
Pu 242	< 1.67 E-03	< 7.02 E-05
Am 241	< 2.88 E-01	< 1.21 E-02
Total α activity	< 3.09 E+00	< 1.30 E-01

Note: After gross α -measurement, nuclide specific activities are deduced using a reference spectrum. It would not be relevant to quote uranium isotopes activities which are far lower than that of plutonium isotopes, as uranium specific activity is negligible compared to plutonium specific activity.

Reference facilities and generic discharges

83. These processes are undertaken either together or separately at a number of sites in Europe. The reference site was taken to be Springfields in the UK as there is an extensive database of discharges of radionuclides from this site. Enrichment is not undertaken at Springfields; however releases from enrichment are trivial compared with those from other stages of the fuel cycle. The assumed annual discharges are given in Table 9.

84. The discharges were normalised to electricity production assuming a reference burn up of 40GWd/t at approximately 4% U-235 and a PWR thermal efficiency of 30%, together with a correction for the processing of natural uranium at the site.

Table 9: Aerial and liquid discharges from fuel conversion, enrichment and fabrication

Radionuclide	Discharges (GBq y ⁻¹)	
	Aerial	Liquid
Th-230	ND	5.20 10 ¹
Th-232	ND	1.4
Th-234	ND	1.34 10 ⁵
U-234	5.7 10 ⁻¹	5.50 10 ¹

Note: ND - Denotes not discharged

3.3 Power Generation

85. In 1998 some 345 nuclear plants were in operation in the OECD area. There are a number of plants loaded with MOX fuel and this trend is expected to continue grow. Since radioactive discharges/releases are influenced by national regulatory requirements, site specific conditions and plant characteristics, the French PWRs of 900 Mwe series are selected to have meaningful reference data on radioactive discharges. St. Laurent 1 is the first in the series loaded with MOX fuel in 1987 and currently 9 plants are loaded MOX fuel. French experience also allows a comparison of radioactive discharges/releases from the same size plant with and without MOX fuel.

86. The average content of Plutonium in MOX fuel is limited to 5.3 %. At equilibrium, the MOX fuel elements represent approximately 30 % of the total number of the fuel elements in the core.

Typical discharges/releases

87. Comparison of radioactive discharge/release data given in the Annex A demonstrates that use of MOX fuel in reactor did not modify the level of radioactive discharges/releases and the isotopic composition of discharges/releases. For this study, normalised 1996 data per TWh are used on the basis of radioactive discharges/releases data for 900 Mwe series.

88. Normalised liquid and gaseous discharges/releases are given in Table 10.

Table 10: Discharges from PWRs, normalised to reactor capacity

	900 MWe	1300 MWe
liquid discharges		
ΣR.E*	2.5 GBq.(GWe.year) ⁻¹	1.3 GBq.(GWe.year) ⁻¹
tritium	14.7 TBq.(GWe.year) ⁻¹	17.1 TBq.(GWe.year) ⁻¹
gaseous discharges		
gas	<7.7 TBq.(GWe.year) ⁻¹	
halogens and aerosols	<78 MBq.(GWe.year) ⁻¹	

* sum of the γ radionuclide emitters identified

89. It should be noted that releases of C-14 from French reactors are not measured systematically. They are measured at a limited number of plants. On average, the atmospheric discharges of German

PWRs are estimated at 0.1 TBq/GWey and those of BWR are estimated at 0.47 TBq/GWey. Releases of C-14 in liquid discharges is assumed to represent only a tiny part of the total releases of C-14 (approximately 5%). Releases of C-14 from French reactors should be similar.

Reference power production facility and generic discharges

90. The most common nuclear power plant in Western Europe is the Pressurised Water Reactor (PWR) and this type was assumed in this study. Annual discharges from a typical 1300MW reactor were derived on the basis of French data and are given in Table 11. On the basis of the limited available information, it was assumed that there was no significant difference between discharges from a reactor loaded with UO₂ and one loaded with MOX.

91. An electricity generation of 1.07 GWa was taken in normalising the discharges (Depres, personal communication).

Table 11: Aerial and liquid discharges from a typical PWR

Radionuclide	Discharges (GBq y ⁻¹)	
	Aerial	Liquid
H-3	9.00 10 ²	1.75 10 ⁴
C-14	2.15 10 ²	1.61 10 ¹
Ar-41	3.50 10 ¹	ND
Mn-54	ND	1.50 10 ⁻²
Co-58	1.70 10 ⁻⁴	3.65 10 ⁻¹
Co-60	6.50 10 ⁻⁶	1.65 10 ⁻¹
Ni-63	ND	3.96 10 ⁻¹
Kr-85	6.5	ND
Kr-88	2.30 10 ⁻¹	ND
Ag-110m	ND	9.50 10 ⁻²
Sb-124	ND	5.00 10 ⁻²
I-131	1.55 10 ⁻²	1.50 10 ⁻²
I-133	2.00 10 ⁻³	ND
Xe-133	5.0	ND
Cs-134	ND	6.00 10 ⁻²
Cs-137	ND	1.75 10 ⁻²

Note: ND - Denotes not discharged

3.4 Interim Storage and Conditioning of Spent Fuel

Storage facilities for spent nuclear fuel assemblies

92. Irradiated fuel assemblies are stored at reactor sites (AR) or away from reactors (AFR) at reprocessing facilities or separate storage locations. Due to the fact that world-wide no repository for disposal of spent nuclear fuel or high level radioactive waste is in operation, and only a fraction of spent fuel is going for reprocessing, the main share of spent fuel produced up to now is kept in

interim storage. At present the amount of spent fuel assemblies in interim storage is estimated to reach 100 000 t HM. Therefore interim storage has to be acknowledged for the time being and the future as an important step in the nuclear fuel cycle. On the other hand it has to be emphasised that interim storage offers no definite solution to waste management. In any case a repository for the disposal of wastes is needed whether a closed fuel cycle or a once-through fuel mode is pursued.

93. Storage in water pools is the common practice for AR storage after unloading the fuel from the reactor core. This practice is part of reactor operation and covered by the corresponding licence. Therefore radiation exposure of plant personnel and discharges have to be within the operation authorisation of the nuclear power station. The same is valid for the large wet storage pools at reprocessing plants.

94. AFR spent fuel storage has been implemented in several countries as wet storage in pools or as dry storage using concrete canisters, metal casks or concrete vaults. Storage of spent fuel assemblies in water pools is a long-standing mature technology with an excellent safety record. The main safety aspects of cooling, subcriticality, shielding, structural integrity and prevention of corrosion are covered by design and operational procedures.

95. Examples for AFR wet storage are facilities for spent RBMK fuel assemblies in Russia and the underground pools for LWR fuel assemblies CLAB in Sweden and Olkiluoto in Finland. As it became evident that in future large quantities of aged spent fuel with lower heat generation would have to be stored for long time periods various forms of dry storage have been developed. Roughly three forms of dry storage concepts can be distinguished: metal casks, concrete containers or concrete vaults. An overview of AFR storage capacities and of dry storage concepts and their implementation is given in the Annex A.

Discharges

Discharges from AFR pool storage facilities

96. Practical experience from the Olkiluoto and CLAB AFR storage pools shows that the discharges of radioactive substances to the environment are very small. Nuclide-specific discharge data to air and water are available from CLAB for 1996 (see Annex A).

97. In relation to the comparison of different fuel cycle strategies the radiological impact of discharges from AFR wet storage facilities on the population is negligible.

98. Due to the permanent purification of the pool water spent ion-exchange resins contaminated predominately by Co-60 have to be treated for interim storage and disposal.

Discharges from AFR dry storage facilities

99. No discharges of radioactive substances requiring emission control occur at dry cask storage facilities. During the licensing procedure for the German facilities at Ahaus and Gorleben an assessment has been performed on potential activation of air, dust, moisture and construction material due to the very weak neutron emission from the casks. The results confirmed that no specific precautions are necessary in this respect.

100. For the dry interim storage facility for VVER fuel assemblies at Paks, Hungary, designed basis radioactive discharges have been assessed in the safety case (see Annex A).

101. In summary dry storage facilities for spent fuel assemblies show no or only very small discharges of radioactive substances to the environment. For a comparison of fuel cycle strategies this aspect has no relevance.

Conditioning of spent fuel assemblies for disposal

102. In case of a once-through fuel strategy irradiated fuel assemblies have to be packed or conditioned for final disposal after the period of wet or dry interim storage. This handling step has not yet been performed because containers for disposal of fuel assemblies in deep geologic formations are still in the stage of development and no repository is operable. Different modes of packaging and conditioning have been proposed and are under development. The easiest procedure consists just in packaging the complete fuel assemblies in containers suitable for the repository. Other concepts are based on disassembling the fuel bundle to single rods, to rod consolidation or to cutting the rods into few pieces to reduce the length of the container for final disposal.

103. For packaging of intact spent fuel assemblies into containers for disposal, no or only minor radioactive discharges are to be expected. In case of rod consolidation the risk of damage of the rod cladding exists with a potential of the release of volatile radionuclides. For cutting operations Kr-85, H-3 and I-129 are to be released together with small amounts of radioactive aerosols.

104. In the 1987 safety case for the German pilot conditioning plant the discharge data are given as upper limits in the application for a licence (see Annex A). No significant contribution of this step of spent fuel management is expected to the comparison within the scope of this study.

Generic study

105. Discharges from this stage are insignificant and are not considered further in the generic study.

3.5 Reprocessing, Vitrification and Interim Storage

106. There are three reprocessing plants in operation and they are located at La Hague in France, Sellafield in the United Kingdom, Tokai-mura in Japan. The plant at Tokai-mura is a semi-industrial scale plant. A large plant is under construction at Rokkasho-mura in Japan. All plants are capable to reprocess light water reactor fuel but the La Hague plant has the most suitable feature for this study: long and stable operating experience, modern technology. Thus, it is selected as a reference plant for the study.

107. The La Hague plant has two main units: UP2 brought into operation in 1966 was refurbished with new technology and reopened with an increased capacity in 1994 and UP3 started its operation in 1990. 1,670 tHM spent fuel was reprocessed in 1997 and this is equivalent to 397 TWh, assuming a mean burn-up of 30 GWd/t.

108. The spent fuel assembly is mechanically chopped into small pieces and chemically processed to separate Uranium, Plutonium and waste. Uranium and Plutonium are converted in oxide forms. Waste

Table 12. Liquid releases from La Hague, 1997

(In the normalisation the annual throughput of 1670 tHM is assumed to correspond to total electricity production of 76,1 GWh; see text for additional explanation)

Radionuclide	Activity released (GBq/a)	Activity released (GBq/GWh(e)) (76.1 GWh)	Activity released (GBq/GWh(e)) (57.1 GWh)
H-3	$1.19 \cdot 10^{+7}$	$1.56 \cdot 10^{+5}$	$2.08 \cdot 10^{+5}$
C-14	$9.65 \cdot 10^{+3}$	$1.27 \cdot 10^{+2}$	$1.69 \cdot 10^{+2}$
Mn-54	$4.81 \cdot 10^{+1}$	$6.32 \cdot 10^{-1}$	$8.42 \cdot 10^{-1}$
Co-57	$1.37 \cdot 10^{+0}$	$1.80 \cdot 10^{-2}$	$2.40 \cdot 10^{-2}$
Co-58	$1.64 \cdot 10^{+1}$	$2.16 \cdot 10^{-1}$	$2.87 \cdot 10^{-1}$
Co-60	$4.85 \cdot 10^{+2}$	$6.37 \cdot 10^{+0}$	$8.49 \cdot 10^{+0}$
Ni-63	$1.30 \cdot 10^{+2}$	$1.71 \cdot 10^{+0}$	$2.28 \cdot 10^{+0}$
Zn-65	$1.68 \cdot 10^{+0}$	$2.21 \cdot 10^{-2}$	$2.94 \cdot 10^{-2}$
Sr-89	$3.73 \cdot 10^{+1}$	$4.90 \cdot 10^{-1}$	$6.53 \cdot 10^{-1}$
Sr/Y-90	$3.73 \cdot 10^{+3}$	$4.90 \cdot 10^{+1}$	$6.53 \cdot 10^{+1}$
Zr/Nb-95	$3.93 \cdot 10^{-1}$	$5.16 \cdot 10^{-3}$	$6.88 \cdot 10^{-3}$
Tc-99	$1.30 \cdot 10^{+2}$	$1.71 \cdot 10^{+0}$	$2.28 \cdot 10^{+0}$
Ru/Rh-106	$1.96 \cdot 10^{+4}$	$2.58 \cdot 10^{+2}$	$3.43 \cdot 10^{+2}$
Sb-125	$1.34 \cdot 10^{+3}$	$1.76 \cdot 10^{+1}$	$2.35 \cdot 10^{+1}$
I-129	$1.63 \cdot 10^{+3}$	$2.14 \cdot 10^{+1}$	$2.85 \cdot 10^{+1}$
Cs-134	$2.08 \cdot 10^{+2}$	$2.73 \cdot 10^{+0}$	$3.64 \cdot 10^{+0}$
Cs-137	$2.46 \cdot 10^{+3}$	$3.23 \cdot 10^{+1}$	$4.31 \cdot 10^{+1}$
Ce/Pr-144	$2.94 \cdot 10^{+0}$	$3.86 \cdot 10^{-2}$	$5.15 \cdot 10^{-2}$
Eu-154	$4.09 \cdot 10^{+0}$	$5.37 \cdot 10^{-2}$	$7.16 \cdot 10^{-2}$
U	$6.19 \cdot 10^{+0}$	$8.13 \cdot 10^{-2}$	$1.08 \cdot 10^{-1}$
Pu-238	$9.38 \cdot 10^{+0}$	$1.23 \cdot 10^{-1}$	$1.64 \cdot 10^{-1}$
Pu-239/240	$4.97 \cdot 10^{+0}$	$6.53 \cdot 10^{-2}$	$8.70 \cdot 10^{-2}$
Pu-241	$2.09 \cdot 10^{+2}$	$2.75 \cdot 10^{+0}$	$3.66 \cdot 10^{+0}$
Am-241	$5.70 \cdot 10^{+0}$	$7.49 \cdot 10^{-2}$	$9.98 \cdot 10^{-2}$
Cm-244	$2.45 \cdot 10^{+0}$	$3.22 \cdot 10^{-2}$	$4.29 \cdot 10^{-2}$
Other α emitters	$1.90 \cdot 10^{+1}$	$2.50 \cdot 10^{-1}$	$3.33 \cdot 10^{-1}$

Table 13. Gaseous releases from La Hague, 1997

(In the normalisation, the annual throughput of 1670 tHM is assumed to correspond to a total electricity production of 76,1 GWa; see text for additional explanation); 4th column gives the same figures corresponding 57.1 GWa assumption.

Radionuclide	Activity released (GBq/a)	Activity released (GBq/GWa(e)) {76.1 GWa}	Activity released (GBq/GWa(e)) {57.1 GWa}
H-3	$7.57 \cdot 10^{+4}$	$9.95 \cdot 10^{+2}$	$1.33 \cdot 10^{+3}$
C-14	$1.70 \cdot 10^{+4}$	$2.23 \cdot 10^{+2}$	$2.98 \cdot 10^{+2}$
Kr-85	$2.97 \cdot 10^{+8}$	$3.90 \cdot 10^{+6}$	$5.20 \cdot 10^{+6}$
Ru/Rh-106	$3.24 \cdot 10^{-2}$	$4.26 \cdot 10^{-4}$	$5.67 \cdot 10^{-4}$
I-129	$1.67 \cdot 10^{+1}$	$2.19 \cdot 10^{-1}$	$2.92 \cdot 10^{-1}$
I-131 ³	$1.18 \cdot 10^{+0}$	$1.55 \cdot 10^{-2}$	$2.07 \cdot 10^{-2}$
I-133 ¹	$3.11 \cdot 10^{-1}$	$4.09 \cdot 10^{-3}$	$5.45 \cdot 10^{-3}$
Cs-137	$5.96 \cdot 10^{-5}$	$7.83 \cdot 10^{-7}$	$1.04 \cdot 10^{-6}$
Pu-238	$7.46 \cdot 10^{-6}$	$9.80 \cdot 10^{-8}$	$1.31 \cdot 10^{-7}$
Pu-239/240	$5.99 \cdot 10^{-6}$	$7.87 \cdot 10^{-8}$	$1.05 \cdot 10^{-7}$
Other α emitters	$1.83 \cdot 10^{-3}$	$2.40 \cdot 10^{-5}$	$3.20 \cdot 10^{-5}$
Other β emitters	$2.94 \cdot 10^{-2}$	$3.86 \cdot 10^{-4}$	$5.15 \cdot 10^{-4}$

Reference reprocessing facility and generic discharges

112. Commercial reprocessing is undertaken at two sites in Western Europe, the BNFL Sellafield site on the north-west Coast of England and the COGEMA site at La Hague on the north-west coast of France. Oxide fuel of the type used in PWRs has been reprocessed at La Hague for much longer than at Sellafield and there is extensive experience of discharges from La Hague. Therefore the La Hague site was chosen as the reference one for this study. The assumed discharges are given in Tables 12 and 13.

3.6 Decommissioning and Dismantling of Nuclear Facilities

113. Nuclear facility is eventually decommissioned and dismantled. The site may be decontaminated for unrestricted use. MOX fuel fabrication facility and reprocessing facility are additional ones to the one-though option. Radioactive waste generated from the decommissioning and dismantling operation is managed in accordance with the national regulatory requirement.

114. A number of nuclear facilities have been decommissioned and dismantled in the OECD area but they are relatively small facilities. Experience in past decommissioning and dismantling operation shows that radioactive discharges/releases are limited and waste is conditioned in accordance with the national regulatory requirements. Radioactive discharges/releases and waste generated from decommissioning and dismantling of MOX fuel fabrication and reprocessing facilities are considered to be negligible when they are normalised by total TWh of fuel processed.

3 These radionuclides come from Curium spontaneous fission.

3.7 Disposal of Solid Waste

115. Radioactive waste fall into two broad categories: low- and high- level waste. Low-level waste contains small amount of radioactive nuclides with short half-lives. This type of waste is very voluminous. Most waste generated at nuclear facilities fall into this category. High-level waste contain large amount of radioactive nuclides with long half-lives. High-level waste includes vitrified waste from reprocessing plant and encapsulated spent fuel. High-level waste contains more than 99 % of radioactivity generated but its volume is very small. Waste generated at MOX fuel fabrication would require similar long-term consideration to this waste due to Plutonium contamination. Sometimes the low-level waste containing higher amounts of radioactivity is referred to as intermediate-level waste.

116. All waste generated from fuel cycle steps except for mining and milling is sorted out by contamination levels or by nature of waste and conditioned at the site of its generation to facilitate subsequent handling in compliance with the national regulatory requirements. Radioactive waste generated at mining and milling step is disposed of on the site as it is described in the mining and milling section.

117. Low- and intermediate-level waste is disposed of into near-surface repository. Currently preferred option for high-level waste is disposal into deep underground repository. The objective of solid waste disposal is either to provide complete isolation during the decay period or to provide a possibility to defer the possible releases in the far future and distribute the releases for a long release Period.

Characteristics of waste generation of two options

118. In general no-reprocessing option generates more low-level waste than reprocessing option, particularly at mining and milling, conversion, enrichment and fuel fabrication steps. No-reprocessing option does not produce high-level waste but all spent fuel is disposed of in the same manner as the high level waste.

119. Reprocessing option generates more waste contaminated with Plutonium at MOX fuel fabrication and reprocessing steps and high-level waste.

Discharges/releases

120. No discharge is assumed from repositories for low-level waste.

121. The preferred option for disposal of high-level waste, including spent fuel, is to place it in a deep repository in different types of geological formations. Several programmes to develop such repository are underway in the NEA Member countries. Generic studies have shown that long-term isolation of long-lived radioactive waste is feasible. *[Reference to the NEA reports]*

122. The tunnel system of the Finnish project for disposal of spent fuel in crystalline rock is located at a depth of 500 m. In each tunnel several individual disposal holes are bored into the tunnel floor. Each hole houses one canister and is surrounded by buffer material made of clay-like material bentonite. The buffer material prevents ground water from contacting with canister and retarding

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migration of radioactive nuclides into ground water system when the canister integrity is eventually lost. Similar multi-barrier system is proposed by other geological disposal projects.

123. Under the normal situation, radionuclides in canister are allowed to decay to an unarmful level and there will be no significant radioactive discharges/releases from the repository. For this study, it is assumed that there is no radioactive discharge/releases after disposal.

3.8 Transportation

124. Transportation of radioactive materials in the nuclear fuel cycle is a key activity to form a fuel cycle. Transportation of radioactive materials is regulated by national and international regulations which are based on the IAEA Regulations for Safe Transport of Radioactive Materials. In the fuel cycle, natural Uranium, enriched Uranium, Plutonium, fresh fuel assembly, spent fuel assembly and conditioned waste are transported.

125. Transportation of natural Uranium, enriched Uranium, fresh and spent fuel assembly, conditioned waste is common to the two fuel cycle options. Reprocessing option involves transportation of Plutonium and vitrified high level waste in addition to the common segments. The no-reprocessing option requires 20 % more transportation of bulky, low-level radioactive waste. On the other hand, it requires less transportation of high level radioactive materials.

126. Since all materials to be transported are packed in sealed containers, no radioactive release is considered in this study but external doses to workers and to the public are reviewed in Chapter 5. The differences may be brought about, on one hand by the location of installations, on the other hand by factors that are inherent to the systems. These factors include the large amounts of fresh Uranium ore and spent fuel and low- and intermediate level waste in the no-reprocessing option and HEW in the reprocessing option.