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Environmental aspects based on operational performance of nuclear fuel fabrication facilities

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ENVIRONMENTAL ASPECTS BASED ON OPERATIONAL PERFORMANCE OF
NULEAR FUEL FABRICATION FACILITIES

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FOREWORD

This publication was prepared within the framework of the IAEA Project entitled Development and Updating of Guidelines, Databases and Tools for Integrating Comparative Assessment into Energy System Analysis and Policy Making, which included the collection, review and input of data into a database on health and environmental impacts related to operation of nuclear fuel cycle facilities.

The project followed the activities carried out in the context of the DECADES project in the early nineties which resulted in the generic description of health and environmental aspects of nuclear facilities (IAEA-TECDOC-918). The report was limited to the evaluation of collective and in some cases individual average doses and radioactive releases from the facilities. It did not consider non-radiological hazards to the workers and the public resulting from construction, operation and decommissioning of these facilities.

The objectives of the project included assembling environmental data on operational performance of nuclear fuel fabrication facilities in each country; compiling and arranging the data in a database, which will be easily available to experts and the public; and presenting data that may be of value for future environmental assessments of nuclear fuel fabrication facilities.

In this context, several consultancy meetings on fuel fabrication facilities were held in 1996, 1997, and 1999. The publication consists of 8 Sections reflecting environmental aspects of fuel fabrication facilities obtained at the consultancies. The Annex is a collection of some papers presented by the participants.

Appreciation is expressed to all those who participated in the preparation of this publication and also to the Member States that sent experts to assist the IAEA in the work.

The IAEA officers responsible for this publication were Y. Orita, N. Ojima and K. Kawabata of the Division of Nuclear Fuel Cycle and Waste Technology.

EDITORIAL NOTE

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CONTENTS

1. INTRODUCTION

Environmental data on nuclear power plants have been compiled and analysed by various organizations and are well documented. Environmental data on nuclear fuel cycle facilities may be less well documented. This is especially true for data based on operational performance worldwide. However, some publications exist on the radiological impact of the nuclear fuel cycle [1–3].

National authorities generally compile environmental data on nuclear fuel cycle facilities. It is hoped that collecting such data and presenting it in a form, which can be easily understood by both experts and the public, will lead to a greater understanding of nuclear fuel cycle facilities.

In order to make a reliable assessment of the environmental impact of the nuclear fuel cycle as a whole, it is important to collect accurate data even from those facilities, which have little environmental impact. The scope of this study has been limited to nuclear fuel fabrication facilities, including mixed-oxide (MOX) facilities.

2. OVERVIEW

2.1. The different reactor types

Nuclear power reactors produce energy when heavy atoms, primarily uranium and plutonium, fission under the impact of a neutron flux, itself produced by fission.

The various components of the reactor "core" are arranged in a highly precise configuration to initiate the chain reaction, to allow the energy released by the fissile material to be recovered, and to control the fission process. The structural materials support and contain the materials participating in the nuclear reaction. This material consists of:

- fissile (^{235}U, ^{239}Pu, ^{241}Pu) isotopes; and
- fertile (even-numbered U and Pu) isotopes which become fissile when they capture a neutron not involved in the fission process.

The reactor (or reactor series) is defined by the materials it uses, the core configuration, the moderator (in case the neutrons need to be slowed down to thermal energy) and the coolant. After many attempts to commercialize a wide range of reactor types, only a few have been selected around the world to generate power (for a more detailed description of the various reactor types, see for example [4–6]):

- gas cooled, graphite-moderated reactors (GCR or Magnox with U-metal and advanced gas cooled reactors, AGR, with UO_2 fuel);
- reactors cooled and moderated with light water (LWRs), of which there are two types: pressurized water reactors (PWRs, including the Russian version, WWER) and boiling water reactors (BWRs);
- reactors cooled (under pressure) and moderated with heavy water (Canada's CANDU reactor or PHWR or HWR);
- reactors cooled with light water but moderated with a different element: graphite (LWGRs or RBMKs, in Russian) and heavy water (the Japanese advanced thermal reactor, ATR);
- not moderated reactors or fast breeder reactors (FBRs or LMFBRs), which are cooled with liquid metal (sodium).

2.2. Nuclear fuel cycle activities and options

2.2.1. Nuclear fuel cycle activities

The nuclear fuel cycle comprises those activities, which are required to produce suitable fissile material for the operation of the various nuclear power reactors (i.e. front end fuel cycle activities) and those to manage and dispose the used material (i.e. back end fuel cycle activities).

The main front end fuel activities comprise:

- uranium mining and milling;
- uranium refining and conversion;
- uranium enrichment and conversion;
- fuel fabrication.

The main back end fuel cycle activities comprise:

- spent fuel storage;
- spent fuel treatment (for disposal);
- spent fuel reprocessing;
- waste treatment (for disposal).

Between the various main activities, some minor activities are required, such as treatment, transport and storage of the respective end products. See [4–7] for a more detailed description of the nuclear fuel cycle activities.

2.2.2. Nuclear fuel cycle options

The management of the used material, i.e. the spent fuel management, allows to choose between a few fuel cycle options. Once the spent fuel has been discharged from the power reactor, it can be directly disposed of (open or once-through fuel cycle) or reprocessed (closed fuel cycle) to recover the remaining fissile products representing about 30% (for LWR) of the energy initially contained.

Disposal in a deep geological repository involves steps, which would place the spent fuel in a location under conditions which would not generally allow for its removal. Reprocessing allows for the separation of the fissile plutonium and uranium from the waste materials, which are conditioned in a form suitable for long term storage and final disposal.

Originally, the intention behind the closed fuel cycle concept was to recycle the separated plutonium and uranium in breeder reactors. However, owing to delays and cancellations of breeder programmes, the separated fissile materials are being recycled in thermal reactors. At present, thermal recycling of plutonium as mixed oxide (MOX) fuel in LWRs is being carried out in Belgium, France, Germany, Switzerland and Japan. Thermal recycling of uranium is being carried out in the RBMKs in the Russian Federation and in the AGRs in the UK.

Another approach is the deferral of the decision to choose between the open or closed fuel cycle and is called the “wait and see “ strategy resulting in the decision to only consider interim storage of the spent fuel. This approach provides the ability to monitor the storage continuously and to retrieve the spent fuel later for either direct disposal or reprocessing. Several countries with nuclear programmes are using this approach.

Countries can be divided broadly into the following three groups according to their current spent fuel management policies (some countries are following more than one approach):

- Countries which have selected the once-through fuel cycle followed by disposal of the spent fuel;
- Countries which have selected the closed fuel cycle and are operating or constructing reprocessing plants or have contracts for reprocessing abroad;
- Countries that have deferred their decision and are still evaluating their spent fuel management options.

There is a scientific consensus that all three of these spent fuel management strategies provide adequate protection to the public and the environment.

3. STATUS OF FUEL FABRICATION AND MARKET OVERVIEW

Uranium oxide fuels serve the main reactor programmes. Fuel elements can be made from natural uranium, such as in CANDU and Magnox reactors, or from enriched uranium, as in LWRs and AGRs. Plutonium fuel was initially developed for the FBR programmes in France, Japan, Kazakhstan and the Russian Federation. Currently, it is increasingly used in LWRs as MOX fuel in several countries.

Table I summarizes the present status of uranium fuel fabrication facilities in the various countries [8]. In 2000, fuel for commercial nuclear reactors was fabricated in 18 countries. These countries have a global fabrication capacity of 19 676 t U/a. Table II shows the available fuel fabrication capacities in Member States according to their end product, i.e. fuel types used in power reactors. The present status of MOX fuel fabrication facilities in the various countries is given in Table III [8]. The use of MOX fuel in LWRs in some countries (France, Belgium, Switzerland, Germany and Japan) will require additional capacity over the global fabrication capacity of 232 t HM/a [9].

IAEA estimates of fuel requirements are given in Table IV by reactor type. The Table shows that there is already an important over-capacity in fuel fabrication and the forecast is for this over-capacity to increase significantly with time. Frozen nuclear programmes in many countries and decommissioning of older plants, hardly offset by new installations in few countries, can explain this over-capacity. Fuel fabricators will have to adapt to these changing patterns in an effort to find through alliances, mergers, aggressive marketing and internationalization of their operations a place in what is likely to be an extremely competitive business.

TABLE I. STATUS OF URANIUM FUEL FABRICATION FACILITIES IN THE WORLD[a]

COUNTRY	Facility	Fuel Type/Process	Start of Operation	Shutdown	Capacity t HM/a	Status
Argentina	Nuclear Fuel Manufacture Plant	PHWR	1982		160	**In operation**
Belgium	FBFC International	BWR, PWR	1961		500	**In operation**
Brazil	Resende - Unit 1	PWR	1982		120	**In operation**
Canada	Moncton	PHWR		1986	250	**Decommissioned**
	Peterborough Facility	PHWR (Bundles)	1956		1200	**In operation**
	N. Fuel PLLT. OP. - Toronto	PHWR (Pellets)	1967		1300	**In operation**
	Chalk River Laboratories, NFFF	LEU and HEU	1990		1	**In operation**
	Zircatec Precision Ind.	PHWR	1964		1500	**In operation**
China	Yibin	PWR	1993		100	**In operation**
	Candu Fuel Plant	PHWR	2003		200	**Under construction**
Korea, Rep. of	Nuclear Fuel Fabrication Plant		1992		0	**Stand by**
France	FBFC - Romans	Dry for PWR (U and Rep. U)	1979		800	**In operation**
	SICN	FBR (Blanket, UO2)	1960		150	**In operation**
	FBFC - Pierrelatte	PWR	1986	1999	500	**Shutdown**
	SICN	NUGG (GCR)	1957	1992	1000	**Stand by**
Germany	Hobeg Fuel Fabrication Plant	HTR, THTR	1972	1989	4	**Decommissioned**
	Siemens Brennelement. Karlstein	LWR	1966	1995	40	**Decommissioned**
	Nukem GMBH	MTR, HTR	1960	1988	100	**Decommissioning**
	Siemens Brennelement. Hanau	BWR, PWR	1969	1995	750	**Decommissioning**
	Lingen	BWR, PWR	1979		650	**In operation**
India	NFC - Hyderabad (BWR)	BWR	1974		25	**In operation**
	New U Oxide Fuel Plant-1	PHWR (Pellets)	1998		300	**In operation**
	NFC - Hyderabad (PHWR)	PHWR	1974		300	**In operation**
	Trombay	PHWR	1982		135	**In operation**
	New U Fuel Assembly Plant-1	PHWR	1994		600	**Under construction**
Italy	CONU Magnox Fuel Fabr. Plant		1960	1987		**Decommissioned**
	Fabricazioni Nucleari SPA	PWR	1974	1990	200	**Stand by**
Japan	Japan Nuclear Fuel Co. Ltd. (JNF)	BWR	1970		850	**In operation**
	Nuclear Fuel Industry Ltd. (NFI Tokai)	BWR	1980		200	**In operation**
	Mitsubishi Nuclear Fuel Ltd. (MNF)	PWR	1972		440	**In operation**
	Nuclear Fuel Industry Ltd. (NFI Kumatori)	PWR	1972		284	**In operation**
Kazakhstan	UST - Kamenogorsk	RBMK			2000	**In operation**
Korea, Rep. of	Candu Fuel Fabrication Plant	PHWR	1987		400	**In operation**
	Korea Nuclear Fuel Co. Ltd.	PWR	1989		400	**In operation**
Pakistan	Chashma	PHWR	1986		20	**In operation**
Romania	Pitesti Fuel Fabrication Plant (FCN)	PHWR	1983		150	**In operation**
Russian Fed.	Machine - Building Plant (WWER)	WWER-440, WWER-1000			620	**In operation**
	Machine - Building Plant	UF6 / UO2 for LWR	1996		800	**In operation**
	Machine - Building Plant (FBR)	FBR			50	**In operation**
	Novosibirsk	WWER-1000	1949		1000	**In operation**
	Machine - Building Plant (RBMK)	RBMK	1961		900	**In operation**
	Novosibirsk	WWER-1000	2000		100	**Planned**
South Africa	Beva	PWR	1988	1996	100	**Shutdown**
Spain	Fabrica de Combustibles	Fuel for PWR and BWR	1985		300	**In operation**
Sweden	Westinghouse Fuel Fabrication Plant	BWR, PWR	1971		600	**In operation**
United Kingdom	Dry Recovery Plant	Enriched Uranium Recovery	1974	1975		**Decommissioned**
	B209S Dry Granulation Production	Dry granulation production		1988		**Decommissioning**
	Springfilds OFC LWR Line	UO2 Pellets / LWR/ WWER-440	1996		330	**In operation**
	Springfields Magnox Canning Pl.	U (Metal) / MAGNOX	1960		1300	**In operation**
	Springfilds OFC AGR Line	UF6 / UO2 / (IDR) / AGR	1996		290	**In operation**
	UKAEA Fuel Fabrication Plant	MTR	1958		(500)b	**In operation**
	Springfields (PWR)	PWR	1984	1995	300	**Shutdown**
	A58 Pellet Plant	UO2 / LWR Pellets	1985	1998	150	**Shutdown**
	Springfields AGR Canning Pl.	AGR	1968	1999	280	**Shutdown**
USA	Fuel Fabrication Facility Attleboro	Uranium fuel elements	1952	1981		**Decommissioned**
	Uranium Fuel Fabrication Plant	Slightly enr. uranium fuel	1957	1966		**Decommissioned**
	U Fuel Fabr. Plant - San Jose		1962	1978		**Decommissioned**
	LANL TA-21	Enr. uranium processing	1950	1984		**Decommissioned**
	Uranium Man. Fac. (Compton)	Depleted uranium	1967	1986		**Decommissioned**
	Apollo	FBR, PWR, MOX	1957	1986	360	**Decommissioned**
	RMI Extrusion Plant	Depleted U	1962	1988		**Decommissioning**
	DP West Plutonium Facility	Plutonium Metal	1944	1978		**Decommissioning**
	FC Fuels	PWR	1982		400	**In operation**
	Richland (ANF)	BWR, PWR	1970		700	**In operation**
	ABB-CE	LWR	1986		450	**In operation**
	Columbia (Westinghouse)	PWR	1986		1150	**In operation**
	Wilmington	BWR	1982		1200	**In operation**
Totals	**In operation**	LWR fuel			9500	**In operation**
		PHWR fuel			3865	**In operation**
		RBMK, WWER fuel			4520	**In operation**
		GCR, Magnox, AGR fuel			1590	**In operation**
		FBR fuel, other			201	**In operation**
		All types of fuel fabrication			19676	**In operation**
	Stand by	LWR fuel			200	**Stand by**
		GCR fuel			1000	**Stand by**
	Under construction	PHWR fuel			800	**Under construction**
	Planned	WWER fuel			100	**Planned**

[a] Commercial facilities; in addition, there are 10 pilot plants and 5 laboratory scale facilities in the data base. (Source NFCIS, [8])
[b] elements/a

TABLE II. DISTRIBUTION OF URANIUM FUEL FABRICATION CAPACITIES ACCORDING FUEL TYPE IN 2000

(t HM/a)

Country	LWR	PHWR	RBMK	GCR	AGR	Other
Argentina		160				
Belgium	500					
Brazil	120					
Canada	1	2 700				
China	100					
France	800					150a)
Germany	650					
India	25	435				
Japan	1 774					
Kazakhstan			2 000			
Korea, Rep. of	400	400				
Pakistan		20				
Romania		150				
Russian Fed.	1 620		900			50b)
Spain	300					
Sweden	600					
UK	330			1 300	290	500c)
USA	3 900					
Total	11 120	3 865	2 900	1 300	290	201

(a) Blanket fuel for FBR
(b) FBR fuel
(c) assemblies/a for material testing reactor (MTR)

(Source: IAEA NFCIS, [8])

In the context of global liberalization of trade and electricity generation, electric utilities will be induced to shift their traditional procurement policies into more aggressive ones and this will have an impact on the less efficient nuclear fuel suppliers which will be forced to abandon the market. In spite of economic considerations, some countries will maintain their fuel fabrication installations for strategic and energy policy reasons.

Fuel fabrication context in the United States and Europe is significantly different. In the United States the fabrication market is large and consolidated, in spite of the lack of new plants, and has become extraordinarily competitive for the few companies operating in this business. Fabrication costs and prices are much lower than in Europe. However, there are indications that the European fabrication prices (which differ considerably among countries) will be reduced as a consequence of a greater liberalization of the market. In this sense, the signing of the U.S. — EURATOM bilateral agreement will play a major role in the process of weakening the protective barriers and forcing fuel fabricators to compete internationally.

Uranium fuel requirements worldwide in 2000 were about 8 800 t HM, which is equivalent to about 45% of the total fabrication capacity (Table IV). Nevertheless, a number of countries are embarking on national programmes to set up domestic capabilities for reactor fuel fabrication. These include both countries with established nuclear power programmes (e.g. China, the Republic of Korea, Mexico, Romania and Ukraine) and those planning to begin nuclear programmes (e.g. Turkey and Indonesia). The over-capacity shown in Table IV is mostly attributable to LWR fuel fabrication, which has more than twice the capacity required.

The projected fuel fabrication requirements, calculated using the IAEA low rate of growth case for nuclear capacity installed (an increase of around 0.9%/a), show an average decrease of roughly 2.4%/a to nearly 6 900 t U/a in 2010. This evolution is mainly due to the shutdown of GCR type reactors and the expected burnup increase. Although the situation continues to deteriorate, the trend to install additional fuel fabrication facilities to satisfy national requirements seems to be stronger than the alternative of buying fuel fabrication services on the open market. This development will increase pressure on the fuel market unless new nuclear capacity is constructed.

TABLE III. STATUS OF MOX FUEL FABRICATION FACILITIES IN THE WORLD[a]

COUNTRY	Facility	Fuel Type/Process	Start of Operation	Shutdown	Capacity t HM/a	Status
Belgium	Belgonucléaire PO Plant	MIMAS Process for LWR MOX	**1973**		40	In operation
Belgium/France	New Joint Facility	MOX	**(2005)**		50	Planned
China	Demo Facility	MOX				Planned
France	Melox	Advanced-MIMAS for LWR	**1995**		100	In operation
	Cogema - CFCa	MIMAS for LWR, COCA for FBR	**1961**		40	In operation
Germany	ALKEM Fuel Fabrication Plant	MOX	**1965**	1972	8	Decommissioned
	Siemens (MOX fuel), demo	FBR, LWR MOX	**1963**	1992	30	Decommissioning
	Siemens (MOX fuel)	BWR / PWR MOX			120	Deferred
India	AFFF	MOX	**1993**		50	In operation
Italy	Plutonium Lab.	MOX	**1968**	1987	1	Decommissioned
Japan	Takeyama	MOX		1979	10	Decommissioned
	JNC Tokai, PFDF	MOX	**1965**		0.3	In operation
	JNC Tokai, PFFF	FBR	**1972**	1988	1	Shut down
	JNC Tokai, PFFF	ATR	**1972**		10	In operation
	JNC Tokai, PFPF	FBR	**1988**		5	In operation
	New facility	MOX	**(2005)**		100	Planned
Russian Federation	RIAR	MOX	**1975**		1	In operation
	Mayak Paket	MOX	**1980**		0.5	In operation
	Krasnoyarsk	WWER-1000			0	Deferred
	Mayak - Complex 300	FBR, WWER			60	Deferred
	DeMOX / ToMOX	MOX	**(2002)**			Planned
United Kingdom	Sellafield MOX Plant (SMP)	LWR MOX			120	Awaiting license
	Sellafield (MDF)	MOX	**1963**	2000	8	Standby
	UKAEA FMC	MOX	**1962**	1992	10	Decommissioned
	BNFL Coprecipitation Plant	Coprecipitation - (Pu, U)O2 powder	**1969**	1976		Decommissioning
	BNFL MOX For FBR	MOX, FBR	**1970**	1988	6	Shutdown
USA	Erwin	MOX	**1965**	1972		Decommissioned
	Pu FF	MOX	**1959**	1973		Decommissioned
	DeMOX / ToMOX	MOX	**(2005)**		0	Planned
Totals	In operation	LWR			232	In operation
		FBR			5	In operation
	Standby/Awaiting license	LWR			128	Standby/Aw. license
	Planned	LWR			>150	Planned

[a] Commercial and pilot plants; in addition, there are 6 laboratory scale facilities in the data base (Source: NFCIS [8])

TABLE IV. IAEA CURRENT PROJECTED FUEL REQUIREMENTS

(t HM/a)

Reactor type	Country	2000	2005	2010
LWR	Armenia	23	21	13
	Belgium	86	55	53
	Brazil	29	45	43
	Bulgaria	87	75	26
	China	33	52	50
	Czech Republic	75	70	52
	Finland	41	52	35
	France	978	944	933
	Germany	358	329	269
	Hungary	40	38	28
	India	8	-	-
	Indonesia	-	-	17
	Japan	693	650	659
	Korea, D.P.R.	-	15	30
	Korea, Rep. of	147	149	169
	Mexico	22	21	30
	Netherlands	7	-	-
	Philippines	10	9	9
	Russian Fed.	185	200	205
	Slovakia	86	80	75
	Slovenia	10	9	9
	South Africa	29	26	25
	Spain	112	95	91
	Sweden	163	137	47
	Switzerland	48	40	15
	Ukraine	232	236	220
	UK	18	17	16
	USA	1678	1592	1377
Total	Requirementsa	5293	5060	4666
	Capacity	11120	11220	11220
	Over-capacity	5827	5960	6354
PHWR	Argentina	168	95	75
	Canada	1620	1152	1009
	India	292	256	256
	Korea, Rep. of	65	94	112
	Pakistan	13	-	-
	Romania	129	92	73
Total	Requirements	2287	1689	1525
	Capacity	3865	4665	4465
	Over-capacity	1578	1376	1540
RBMK	Lithuania	155	195	195
	Russian Federation	435	390	265
	Ukraine	75	75	-
Total	Requirements	665	660	460
	Capacity	2900	2900	2900
	Over-capacity	2235	2240	2440
AGR	UK	259	259	216
GCR	UK	305	101	-

[a] The LWR requirement totals include values for Taiwan, China, of 95, 103 and 98 t HM/a for the years 2000, 2005 and 2010.

4. REACTOR FUEL TECHNOLOGY

Reactor fuel consists of reactive material, containing uranium or plutonium and a metal sheet or cladding to protect this material from its environment and to prevent fission products from escaping. Currently, one distinguishes mainly between uranium metal and uranium oxide fuel. Metal fuel is used in the form of massive rods and handled individually. Oxide fuel is formed in thinner rods because of its lower conductivity. These rods are assembled into clusters, assemblies, bundles or elements for use in the various reactor types.

The main stages of fuel fabrication consist of purification of uranium, conversion to hexafluoride (if enrichment is needed), conversion to the required chemical form (metal or oxide), formation into the required shape, application of cladding, building into clusters (for oxide) and addition of end-fittings.

For the purposes of this report, LWR fuel fabrication facilities are taken to include all the operations between the receipt of enriched uranium in the form of UF_6 and the completed fuel element. FBR and PHWR (CANDU) fuel fabrication processes start from uranium dioxide and not from UF_6, since these reactors do not require uranium enrichment. For MOX fuel fabrication facilities, the corresponding operations are those from receipt of PuO_2 and UO_2 powder to the completed fuel element.

4.1. Uranium fuel fabrication

4.1.1. Metal fuel

Metal fuel is used in the Magnox reactors and was used in GCRs. The uranium metal alloy is produced by utilising a highly exothermic, thermite-type reaction between uranium tetrafluoride (UF_4, a dense green powder) and magnesium metal. The UF_4 is intimately mixed with coarse chips of magnesium and compacted into pellets. The metal forming reaction is carried out within a graphite containment held in a sealed stainless steel pressure vessel under an inert atmosphere of argon. The reaction is started by heating to 650°C and is thereafter self-sustaining. It reaches a maximum temperature of over 1 500°C, necessary to reduce the viscosity of the liquid products and so to allow separation. The overall reaction is

$$UF_4 + 2Mg \rightarrow U + 2MgF_2$$

4.1.2. Oxide fuel

The feed material for the manufacture and fabrication of fuel for a LWR is uranium hexafluoride (UF_6) enriched to about 3–5% in uranium-235. The UF_6 is converted to uranium dioxide powder (UO_2) which is formed into pellets, sintered to achieve the desired density and ground to the required dimensions. Fuel pellets are loaded into tubes of zircaloy (a zirconium-tin alloy) or stainless steel, which are sealed at both ends. These fuel rods are spaced in fixed parallel arrays to form reactor fuel assemblies.

The fabrication operations involve the following stages:

- preparation of UO_2 powder of desired size distribution;
- manufacture of UO_2 pellets;
- sintering of the pellets in hydrogen gas;
- grinding of the sintered pellets to the required size;

- washing, drying and loading the fuel pellets into tubes of Zircaloy or stainless steel and sealing the ends with welded caps;
- locating fuel rods in fixed parallel arrays forming the reactor fuel assemblies.

In order to produce UO_2 fuel, it is necessary to convert the UF_6 into UO_2. Three processes have been used to carry this out:

- Reduction of UF_6 to UF_4 with hydrogen followed by hydrolysis of UF_4 with steam according to the reactions:

$$UF_6 + H_2 \rightarrow UF_4 + 2HF$$
$$UF_4 + 2H_2O \rightarrow UO_2 + 4HF$$

These occur in a single integrated kiln — the Integrated Dry Route (IDR).

- Direct conversion of UF_6 into UO_2F_2 followed by precipitation with ammonia to form ammonium diuranate (ADU) and reduction with H_2 to UO_2 :

$$UF_6 + 2H_2O \rightarrow UO_2F_2 + 4HF$$
$$2UO_2F_2 + 6NH_4OH \rightarrow 4NH_4F + (NH_4)_2U_2O_7 + 3H_2O$$
$$(NH_4)_2U_2O_7 + 2H_2 \rightarrow 2UO_2 + 2NH_3 + 3H_2O$$

- The AUC process by which UF_6 is transformed into Ammonium Uranyl Carbonate (AUC) by treatment with CO_2 and NH_3 in water:

$$UF_6 + 3CO_2 + 10NH_3 + 5H_2O \rightarrow (NH_4)_4UO_2(CO_3)_3 + 6NH_4F \quad (500°C)$$
$$H_2 + (NH_4)_4\,UO_2\,(CO_3)_3 \rightarrow UO_2 + 4NH_4 + 3CO_2 + 3H_2O \quad (650°C)$$

A great deal of industrial experience has been gained with these processes in the United States, the United Kingdom, France, Germany and Japan. The main hazard is chemical and is associated with the use of corrosive, flammable and toxic chemicals which also become contaminated with enriched uranium.

The production process for UO_2 pellets involving compaction and sintering has been improved over the years by use of automatic presses, incorporation of shielding and a higher degree of containment than hitherto.\

4.2. MOX fuel fabrication

Two processes are used to produce MOX fuel; these processes differ mainly in the beginning in the nature of the feed materials. Mixed oxide powder is prepared by co-milling (dry process) or co-conversion (wet process), depending on the feed materials. For the dry process, feed materials are UO_2 (AUC, ADU, IDR) and PuO_2 powders. Fabrication of MOX fuel starts with the mechanical mixing of uranium and plutonium oxide powders or by diluting a master blend of UO_2 — PuO_2 to the required enrichment. The alternative wet process starts with Pu- and U –nitrate solutions. Thus plutonium oxide/nitrate is not produced separately. The uranium can be natural or depleted uranium, which is plentiful as a by-product of the enrichment process. The uranium powder has very little radioactivity, whilst the plutonium powder is an alpha emitter, so the process has to be contained for protection.

Further main fabrication steps (pelletizing, sintering, rod fabrication and assembling) are comparable with those of uranium fuel fabrication. The main differences from the UO_2 fuel fabrication are related to the strict alpha activity containment of the processed material in tight glove boxes and the shielding against gamma and neutron radiation. Ageing plutonium may have increasing gamma radioactivity due to the formation of americium.

5. SAFETY, REGULATION PROCEDURE AND STATUS

Like for nuclear power plants, a safety philosophy has to be adopted to ensure high standards of safety in the nuclear fuel cycle [10, 11. 12]. The regulatory regime for nuclear licensing operations varies from Member State to Member State. A general account for the fuel cycle has been described in [13].

5.1. Safety philosophy

All nuclear fuel cycle facilities are subject to regulatory control which requires installations to be designed, constructed, commissioned, operated and decommissioned with due regard to the safety of workers and the general public, as discussed in more detail in the International Basic Safety Standards [12].

These radiation protection standards are based on international understanding of the effects of ionising radiation, and on the recommendations of the International Commission on Radiological Protection (ICRP) [15] which advocated three basic principles:

- "No practice involving exposures to radiation should be adopted unless it produces sufficient benefit to the exposed individuals or to society to offset the radiation detriment it causes;

- In relation to any particular source within a practice, the magnitude of the individual doses, the number of people exposed, and the likelihood of incurring exposures where these are not certain to be received should all be kept as low as reasonably achievable (ALARA), economic and social factors being taken into account. This procedure should be constrained by restrictions on the doses to individuals (dose constraint), or the risks to individuals in the case of potential exposures (risk constraint), so as to limit the inequity likely to result from the inherent economic and social judgements;

- The exposure of individuals resulting from the combination of all the relevant practices should be subject to dose limits, or to some control of risk in the case of potential exposures. These are aimed at ensuring that no individual is exposed to radiation risks that are judged to be unacceptable from these practices in any normal circumstances. Not all sources are susceptible of control by action at the source and it is necessary to specify the sources to be included as relevant before selecting a dose limit."

Releases of radionuclides to the atmosphere or to surface waters from nuclear and other facilities, in which radioactive materials are utilised, are generally strictly controlled in order to protect the health of people living in the local and regional environments. Since the 1970s the IAEA has issued guidance on the control of discharges and, in doing so, it has elaborated upon the basic recommendations of the ICRP. The most recent IAEA guidance on this subject was published in Safety Guide WS-G-2.3 in 2000 [14] This Safety Guide provides practical guidance on the regulation of discharges. For example, it explains procedures for determining whether there is a need for an authorization, and, if required, for determining the most appropriate form and conditions of authorization.

The high standards of safety in nuclear fuel cycle installations are achieved by the nuclear industry because the regulatory bodies require the designers of the facilities to consider not only the need for safety and reliability during normal operation, but also to design against a wide range of potential plant malfunctions including both internal and external hazards. For example, the safety analysis for a nuclear installation could include consideration of explosions, fire, loss of radioactivity containment, criticality, seismic loading, extreme weather conditions and aircraft crash.

In addition to building "defence-in-depth" into the design of a plant, the regulatory bodies also require the facilities to be constructed to the highest standards. Operation of the facility is regularly monitored by the licensing authorities to ensure that operations are carried out in accordance with approved procedures.

5.2. National regulation procedure and status

In this section the role of some national regulators and the effects of the regulatory framework on the safety of nuclear fuel cycle facilities are addressed. The brief descriptions, based on the country reports, are as follows:

5.2.1. France

Regulation procedures start with site selection and plant design and terminate with the dismantling. Nuclear installations are subjected to licensing procedures that include the following steps:

- Plant authorization decree
- Operating licence
- Liquid and gaseous effluent release licenses
- Decommissioning license

Plant authorization decree

Applications for BNI (nuclear material handling facility) authorization decrees are sent to the Minister of Environment and the Minister for Industry, who transmit it to other Ministers (interior, health, transport, town planning, labour,..). Each application comprises a preliminary safety analysis report. Processing of the application includes a public inquiry and a technical assessment.

In view of the recommendations of the standing group of experts, the results of the public inquiry and possible remarks of other Ministers, the Safety Authorities (DSIN) prepare a draft authorization decree. The authorization decree defines the perimeter of the installation and the specific requirements which must be met by the operator.

Operating license

The authorization decree for BNI asks for authorization from the Ministers of Environment and Industry for the facility start up. It is granted after examination by the Safety Authorities of the documents prepared by the operator (comprising the provisional safety analysis report, the general operating rules and the on-site emergency plan). Before an installation is definitely commissioned, the operator must submit a final safety analysis report.

Liquid and gaseous effluent release license

Radioactive effluents and release to the environment are subjected to an administrative license which concerns both their activity and their chemical characteristics. The effluent release license application comprises a description of the operations and their activities, and an assessment of their environmental impact. It is forwarded to the Ministers for Health and for Civil Defense and to the Minister of the Environment. Public and local authorities and organization are consulted by the prefect through a public inquiry and an administrative conference. Authorization is granted by a joint ministerial order signed by the Ministers for Health, Industry and the Environment. This document stipulates:

- The intake and release limits with which the operator must comply;
- The approved method of analysis, measurement and monitoring and of surveillance of the environment;
- The methods to be used for information of the public.

5.2.2. Germany

The Federal Constitution of the Federal Republic of Germany contains detailed provisions on the legislative and administrative competencies of the Federation (Bund) and the individual States (Länder). Pursuant to the *Federal Act of 1959 on the Peaceful Uses of Atomic Energy and Protection Against its Hazards* (Atomic Energy Act), the supreme authorities of the Länder are competent for the granting, withdrawal and revocation of licences for nuclear installations.

The Atomic Energy Act empowers the Federation (Bund) to issue ordinances and general administrative regulations, which are mainly implemented by the Länder acting on behalf of the Federation. The federal control and supervision relate to the legality and expediency of the implementation of the Atomic Energy Act by the Länder. The competent authorities of the Länder are subject to the directives of the competent supreme federal authority, in this case the Federal Ministry for the Environment, Nature Conservation and Nuclear Safety (BMU).

According to the Atomic Energy Act, the construction, operation and possession of nuclear installations are subject to continuous supervision. The supreme authorities of the Länder are responsible for exercising supervisory and control functions, which they may delegate to subordinate agencies, in individual cases. In general, independent experts or expert organizations, namely the Technical Inspection Agencies (TÜV) are involved.

5.2.3. Japan

To license nuclear fuel facilities to run, in each stage (including before construction and before operation), the adequacy of the design, the procedure of construction and the operational procedure and maintenance are examined, in accordance with the *Law for the Regulation of Nuclear Source Material, Nuclear Fuel Material and Reactors* and authorized by the Minister of Economics, Trade and Industry or the Minister of Education, Culture, Sports, Science and Technology (in case of a nuclear fuel material user). In the Law, nuclear fuel facilities are divided into the following: refining facility, fabrication facility, reprocessing facility and nuclear fuel material user. The regulation requirements are different for each category. The uranium commercial scale fuel fabrication facility is prescribed in the category of fabrication facility and existing MOX fuel fabrication facility is prescribed in the category of nuclear fuel material user. Later, commercial scale MOX fuel fabrication facility will be prescribed in the category of fabrication facility.

5.2.4. Spain

The licensing of nuclear fuel fabrication facilities consists of the following laws:

- LAW 25/1964 on nuclear energy (Nuclear Energy Act)

This Act has two main objectives:

Support the development of peaceful applications of nuclear energy in Spain and regulate its practice within the Spanish State.

Protect lives, health and properties against dangers derived from nuclear energy and the harmful effects of ionizing radiation.

- LAW 15/1980 on the creation of the Nuclear Safety Council

This Law creates the nuclear regulation authority, CSN (Consejon de Seguridad Nuclear), as an independent entity from the Central Administration of the State, with its own judicial status as the only competent authority in nuclear safety and radiological protection matters.

- LAW 40/1994 on the regulation of the National Electricity System.

5.2.5. Sweden

The law within the nuclear field is stated in the form of acts and ordinances. The acts are published and written by the Swedish parliament, and the ordinances by the Swedish Radiation Protection Institute (SSI) and Swedish Nuclear Power Inspectorate (SKI). The environmental field is covered by the Swedish Environment Authority. The most important acts and ordinances with relevancy to Westinghouse activities are summarised below.

- The act on Nuclear Activities (SFS 1984:3)
- Ordinance on Nuclear Activities (SFS 1984:14)
- The Radiation Protection Act (SFS1988:220)
- Nature Conservation Act (SFS 1964:822)
- Environment Protection Act (SFS 1969:387)

5.2.6. United Kingdom

In the UK operators of nuclear power plants must, like their counterparts in other industries, conform to the general health and safety standards in the Health and Safety at Work Act 1974 (HSW). Relevant plants (including Springfields) must also comply with the Nuclear Installations (NI) Act 1965 (as amended). The NI Act places an absolute liability on the licensee as regards injury to persons or damage to property arising from a nuclear occurrence without proof of fault on the licensee's part.

HM Nuclear Installations Inspectors are appointed under the HSW Act. They administer the NI Act and deal with nuclear and radiological safety issues at licensed nuclear sites. The Nuclear Installations Inspectorate (NII), which is part of the HSE, monitors and regulates the nuclear and radiological safety aspects of an installation by means of its powers under the HSW Act and the licensing procedures.

Requirements for the protection of the environment and the authorisation of discharges of radioactive waste from nuclear licensed sites are regulated by the Environment Agency (EA), in England and Wales, and by the Scottish Environment Protection Agency in Scotland.

6. ENVIRONMENTAL STATUS OF FUEL FABRICATION FACILITIES

6.1. General

The most significant potential environmental impact results from converting UF_6 to UO_2 and the chemical operations in scrap recovery [1]. Small quantities of radionuclides are released to the environment as gaseous, liquid and solid wastes. Various control systems are employed to reduce the amount of material released offsite and may include scrubbers, demisters, and high efficiency particulate filters for airborne effluents; chemical treatment; onsite burial; and some shipment of solid wastes to commercial burial ground. The annual release of radionuclides from routine operation among the facilities may vary somewhat depending on the types of operation, effluent control systems and operation rates.

Details of the levels of discharge, and representative doses to those members of the public likely to be most exposed, from actual fuel fabrication facilities in some countries are given in the Annex to this report.

The data presented in the Annex is taken from information provided by individual Member States. Most states provide annual reports on environment releases from the nuclear fuel cycle operations. Some states produce annual reports on the levels of discharge from nuclear fuel fabrication facilities specifically and; where available, these are used as the prime source of data. There might be specific requirements relating to the regulatory requirements of individual Member States. If appropriate, commentary on the regulatory regime is included in the appendices.

6.2. Environmental impact

6.2.1. Radiological effluents

The radioactive and non-radioactive emissions of the fuel fabrication facilities represent only a small fraction (about 1‰) of total emissions from the nuclear fuel cycle [16]. Generally working areas in the uranium fuel fabrication facilities and in the MOX fuel facilities are maintained with a negative air pressure to prevent the spread of contamination within the working area or its release outside the working area. The air of these areas is filtered through double cascaded high efficiency particulate air filters and then is discharged through a stack or exhaust outlet. Usually the alpha activity in the exhaust air is continuously monitored. Table V shows the ranges of radioactive air discharges measured in some countries.

In uranium fuel fabrication facilities, wastewater is treated by apparatus to remove radioactive materials. Liquid effluent is collected in a storage tank and measured for the concentrations of radioactivity before being released to the sea or into a river; releases are made only after confirmation that the concentration is within authorization specifications. Table VI shows the range of radioactive water discharges measured in some countries. Liquid waste, which arises from the controlled area in the MOX facilities, is the water from hand washing, the laundry wastewater and the cleaning water from non-contaminated areas. Generally this water is collected in storage tanks and released batch wise into the sea or river, if the radioactive concentration is within authorization specifications.

TABLE V. RADIOACTIVE AIR DISCHARGES

Country	Ref.	Period	Range in GBq
Canada	[16]	1987–1989	≤0.001
China	[3]	<1980[c]	3E-04 - 16
Germany	Annex	1994–1997	1.6E-06 - 1.4E-03
Japan	Annex	1994–1999[a]	8.0E-05 - 7.1E-04
Korea, Rep. of	[16]	1988–1989	0.37 - 0.4
Spain	[17]	1991–1995	3.6E-05 - 8.5E-05
Sweden	Annex	1989–1996	2.1E-03 - 9.2E-03
UK	[16]	1985–1989	2–3
USA[b]	[1]	<1980 [c]	1.4E-03 - 6.3E-02
USA[d]	[1]	<1980 [c]	0.3E-03 - 1.6E-03

[a] Fiscal years from April to March
[b] Including UF_6 conversion
[c] Not indicated in the reference, but prior to 1980
[d] No UF_6 conversion

The amount of radioactive effluent discharged into the environment from nuclear facilities is determined by the operators of the individual facilities and reported to the competent supervising authorities. Details on the extent of measurements, measurement procedures, sampling, instrumentation and documentation of measuring results are often stipulated in the conditions included in the discharge authorization. In many cases, the regulatory authority undertakes additional he measurements for validation purposes.

In general, the actual effluents per year have been far below the licensing values.

TABLE VI. RADIOACTIVE WATER DISCHARGES

Country	Ref.	Period	Range in TBq
Canada	[16]	1987–1989	1.0E-05 - 2.0E-05
China	[3]	<1980[c]	6.9E-04 - 2.91E-02
Germany	Annex	1994–1995	1.5E-04 - 85
Japan	Annex	1994–1999[a]	4.8E-07 - 3.5E-05
Korea, Rep. of	[16]	1988–1989	2.2E-05 - 4.3E-05
Spain	[17]	1991–1995	2.2E-05 - 4.5E-05
Sweden	Annex	1989–1996	1.0E-04 - 8.6E-04
UK	[16]	1985–1989	77.5–161
USA	[1][b]	<1980 [c]	1.6E-04 - 8.1E-02
USA	[1][d]	<1980 [c]	2.3E-05 - 5.9E-05

[a] Fiscal years from April to March
[b] Including UF_6 conversion
[c] Not indicated in the reference, but prior to 1980
[d] No UF_6 conversion

6.2.2. Non-radiological effluents

The level of chemical pollutants in the waste water is controlled by a different system, and appropriate measures are taken to relate levels of chemical pollutants with corresponding limiting values. These measurements may include pH, n-hexane extract, suspended solids (SS), chemical oxygen demand (COD) and based on the pollution control regulation of the local government. The release of waste water is allowed if the concentration of the radioactive material and the pollutants are under the licensing values. The emissions of NO_x, NH_3 and fluoride into the atmosphere are calculated for each stack where necessary.

6.3. Environmental sampling and monitoring

Environmental monitoring programmes have been developed for nuclear fuel cycle facilities in order to verify modelling assumptions and to collect information on the distribution of released radioactive effluents into the environment. These programmes include:

- routine soil sampling for released radionuclides (e.g., uranium). ; Samples are generally taken in the vicinity of the sites, more particularly in the direction of the prevailing wind;
- routine measurement along surface waters and coasts.

The available emergency plans regulate the measurements of radioactive effluents after a possible accidental release. By means of these programmes, immediate and reliable information on the potential distribution of released material is possible. Regular training ensures that in the case of an accident, personnel are acquainted with the measurement programme.

An independent measurement system for the surveillance of the environmental radioactivity is installed in several countries. During normal operation the measurement and information system continuously monitors and documents the radiological background situation in the country and provides early warning in anomalous situations. In emergency situations the system provides initial data on which to base more comprehensive measurements for more detailed assessment purposes.

6.4. Radiation exposure estimation

There are two main forms of dose assessment undertaken in support of the regulatory control of discharges: prospective (before discharge) and retrospective (after discharge) assessments. The former assessment type is one of the factors taken into account in setting a discharge authorization. The IAEA has published guidance on a screening methodology estimating the dose to a group of members of the public likely to represent those most exposed (the critical group). Retrospective dose assessments, based on the results of emission monitoring, provide a method for determining whether public dose criteria have been complied with.

To calculate the total dose it is necessary to take account of contributions from both external irradiation and from intakes of radionuclides, by inhalation or ingestion. Guidance and data needed to perform such calculations is included, for example, in the Safety Report [ref. Safety Report Series No. 19]. For the fuel fabrication facilities external irradiation is generally is not a significant contributor to the critical group dose due to the characteristics of the radionuclides involved.

6.5. Technologies for health and environment

The application of technologies for the protection of health and environment, is determined, in case of establishing nuclear fuel processing plants, by a range of standards related to different safety aspects. Some examples are given below. These may differ depending upon the regulatory structures in place in different countries.

6.5.1. Radiation standards

Radiation protection is based on the recommendations of the ICRP [15], and as expanded in the International Basic Safety Standards [12]. These recommendations and standards apply to the protection of employees and of members of the public [12, 18, 19, 20]. They include general protection principles and annual limits on doses received. More detail on the application of these limits is given in reference [12].

6.5.2. Fire and explosion protection

Prevention of uncontrolled fire and explosion is an important safety goal especially in MOX fuel fabrication, but also in UO_2 fuel fabrication and enrichment. To ensure that this goal will be reached, different types of safety measures are required.

Criteria for design and construction:

- Separation of fabrication and storage sections by fireproof walls;
- As far as possible, use of fire resistant materials for construction of containers and glove boxes;
- Separation of safety systems, ventilation equipment, off-gas filters and burners;
- Separation of supply systems for hydrogen and other burnable media;
- Protection of ventilation systems.

Criteria for operation:

- Stationary and mobile fire fighting systems;
- Automatic fire extinguishing systems for areas with dispersible Pu and Pu storage or with difficult access;
- Water must not be used for fire fighting in areas with criticality risk;
- Strict partitioning of the plant into fire zones, automatic separation of connections (pipes, ventilation system) in case of fire;
- Automatic operation of extinguishing systems (sprinkler, gas), started by fire detectors;
- Fire detectors in glove boxes;
- A specially equipped fire brigade.

6.5.3. Criticality safety

Criticality safety in handling, manufacturing and storage of fissile material is usually based on the double failure principle. In general the proof of criticality safety is based on calculations. For systems with simple geometric shape (cylinder, slab or sphere) data from handbooks may be used. Suitable safety margins must be taken into account. In the criticality handbooks the safety factors, which shall be applied to critical limits, are given [21–23]. In more complicated cases like arrays of fissile material or special geometric shapes criticality safety has to be proven by calculation. Generally a value of $k_{eff} < 0.95$ is required including uncertainties. Neutronic interaction of adjacent components must be checked; interspersed moderation effects are to be taken into account.

To realise criticality safety, essential safety concepts, which are described in detail in national and international standards, are recommended:

- Safe geometry, limitation of dimensions;
- Mass limitation, double batching has to be considered;
- Moderation control, to be applied only for handling of dry material;
- Limitation of enrichment, control by at least two independent ways is required;
- If neutron poisoning is used as part of a safety concept, additional measures for quality assurance are required to ensure the efficacy of the neutron absorber.

Priority should be given to technical safety measures, such as limitation of dimensions instead of administrative ones. A criticality alarm system is required for all areas, where amounts of fissile material are handled or stored, which are large enough to reach criticality under certain conditions. Instructions in case of an alarm have to be provided.

6.6. Energy consumption of facility operation

For a typical stand alone fuel fabrication plant, with an equivalent capacity based on 400 t UO_2/y, the electricity consumption is about 2.5 x 10^4 kWh/tU and the fuel and hydrogen consumption about 3.4 x 10^7 kJ/tU.

7. OPERATOR EXPOSURE

Radiation protection is based on ICRP principles [15]. The protection of employees has to be achieved to a specific standard [20]. For operators in the plant, the individual dose limits are based on external radiation and intakes [19]. In addition to these limits the ALARA principle (as low as reasonably achievable) must be taken into account in design as well as operation. Table VII shows the average and collective whole body exposures of fuel fabrication facilities in some countries.

At uranium fuel fabrication facilities, both the specific radioactivity of uranium and the gamma dose rate in the working environment are low. This means that neither special gamma shielding for the facilities, nor strict containment is necessary, in contrast to the plutonium facility. Workers at uranium fabrication facilities, therefore, receive relatively low occupational doses.

In the MOX fuel fabrication facility, the radiological characteristic of plutonium has to be taken into account. Careful considerations, additional to those of the uranium fuel fabrication facility, must be taken from the viewpoint of the operator exposure. As plutonium has higher specific activity and is more toxic than uranium, MOX fuel must be handled in the glove box. This leads to more constraining conditions on the facility's operability and also requires longer time for the inspection and the maintenance of equipment. The spontaneous fission of plutonium and the alpha-neutron reaction gives rise to higher neutron doses and the co-existent ^{241}Am in the fabrication process, which is the decay product of ^{241}Pu, increases gamma exposure. In the future, with the progressive utilization of recycled plutonium which contains higher isotopic plutonium, the dose in the working environment will become a major concern. To keep the doses low, MOX fabrication processes will need to be designed to be simple, remotely-manipulated and automatically controlled as much as possible.

TABLE VII. OPERATOR EXPOSURE

Country	Ref.	Period	Average	Collective whole body
			mSv	mmanSv
Canada	[24]	1987–1996	2.1–4.26	515a
Germany	Annex	1991–1998	0.38–0.77	101–228
France[b]	[25]	1989–1994	0.7–2.0	12–36
India	[3]	<1981[c]	[d]	450
India	[26]	1992–1997	1–1.6	600–1100
Japan	Annex	1994–1999[e]	0.1–0.4	11–1002
Spain	[17]	1991–1995	0.4–1.25	41–99
Sweden	Annex	1992–1996	0.6–0.78	31–63
UK	Annex	1993–1998	0.85–1.5	1700–4300

[a] 1996
[b] All fuel cycle activities of COGEMA
[c] Not indicated in the reference, but prior to1981
[d] Not indicated in the reference
[e] Fiscal years from April to March

8. CONCLUSIONS

Radiological releases from normal operations of fuel fabrication facilities are a relatively minor contributor to those from the nuclear fuel cycle as a whole. The information included in the Annex suggest that the resultant doses to members of the public, from both uranium and for MOX fabrication facilities, are relatively low.

Doses to workers at fuel fabrication facilities are comparatively low, especially in relation to the dose limit for occupational radiation workers. This experience relates largely to uranium fabrication.Occupational doses at MOX fabrication facilities are potentially larger.

Fuel fabrication facilities-especially fabrication of uranium-contribute only a very small portion of the overall risk associated with operation of the nuclear fuel cycle for electricity generation.

IAEA PUBLICATIONS RELATED TO ENVIRONMENTAL CONSIDERATION ON NUCLEAR FUEL CYCLE FACILITIES

1993 Radioactive Waste Management Glossary

1993 The Safety of Nuclear Installations
(Safety Series No. 110)

1996 Defence in Depth in Nuclear Safety
(INSAG Series No. 10)

1996 Health and environmental aspects of nuclear fuel cycle facilities
(IAEA-TECDOC-918)

1996 International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources
(Safety Standards Series No. 115)

1996 The Nuclear Fuel Cycle Information System and NFCIS CD-ROM and Internet versions (2001).

1999 Occupational Radiation Protection
(Safety Standards Series No. RS-G-1.1)

1999 Status and Trends in Spent fuel Reprocessing
(IAEA-TECDOC-1103)

2000 IAEA Safety Glossary, Terminology used in Nuclear Radiation, Radioactive Waste and Transport Safety, v.1 Vienna Internet version.

2000 Regulatory Control of Radioactive Discharges to the Environment
(Safety Standards Series No. WS-G-2.3)

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ANNEX

GERMANY

1. PRESENT STATUS OF FUEL FABRICATION FACILITIES

1.1. Lingen Advanced Nuclear Fuel GmbH (ANF)

The fuel fabrication plant of ADVANCED NUCLEAR FUELS GmbH is a German subsidiary of the SIEMENS POWER CORPORATION, Bellevue, Washington, USA, which has operations worldwide. Founded in 1975, the plant commenced operation in January 1979 following the approval of a nuclear license and employs a staff of around 300 people.

ADVANCED NUCLEAR FUELS GmbH processes low enriched uranium dioxide (UO_2) with a maximum ^{235}U enrichment of 5 % to produce fuel elements for pressurised water reactors and boiling water reactors. The original scope of production of the plant, encompassing fuel rod and fuel element assembly, was expanded in 1988 to include production of UO_2 pellets. Construction of a UF_6 conversion facility commenced at the end of 1991 in order to complete the production process in operation at the plant.

To fabricate fuel elements, the UF_6 has to be converted chemically to uranium dioxide (UO_2). To achieve this, ADVANCED NUCLEAR FUELS GmbH uses a method of dry conversion of UF_6 developed and patented by SIEMENS POWER CORPORATION. The UO_2 powder produced by dry conversion has the same characteristics as powder manufactured by other producers using wet-chemical conversion processes. The dry conversion process produces no waste or emissions, thus causing no damage to the environment. The fuel fabrication plant of ADVANCED NUCLEAR FUELS GmbH is integrated into the national and international nuclear fuel cycle, the capacity is 400 t/y for UO_2 powder and 650 t/y for fuel element production.

1.2. Siemens Fuel Element Fabrication Plant Hanau, Uranium and MOX

The fabrication of uranium fuel elements with a maximum share of 5 weight-% ^{235}U and the fabrication of mixed oxide fuel elements based on $(U/Pu)O_2$, PuO_2 or UO_2 fuel mainly used in light water reactors is out of operation. The operator plans the decommissioning of both plants for fuel element production at Hanau site. In the future, Siemens will concentrate its uranium fuel manufacturing in the facilities at Lingen (Germany) and Richland (USA).

2. REGULATION PROCEDURE AND STATUS

As indicated by its name, the Federal Republic of Germany is a federal state. The Federal Constitution therefore contains detailed provisions on the legislative and administrative competencies of the Federation (Bund) and the individual States (Länder). Pursuant to the *Federal Act of 1959 on the Peaceful Uses of Atomic Energy and Protection Against its Hazards* (Atomic Energy Act), the supreme authorities of the Länder, designated by their governments, are competent for the granting, withdrawal and revocation of licences for nuclear installations.

The Atomic Energy Act empowers the Federation (Bund) to issue ordinances and general administrative regulations which are mainly implemented by the Länder acting on behalf of the Federation. The federal control and supervision relate to the legality and expediency of the

implementation of the Atomic Energy Act by the Länder. The competent authorities of the Länder are subject to the directives of the competent supreme federal authority, in this case the Federal Ministry for the Environment, Nature Conservation and Nuclear Safety (BMU).

According to the Atomic Energy Act, the construction, operation and possession of nuclear installations are subject to continuous supervision. The supreme authorities of the Länder are responsible for exercising supervisory and control functions, which they may delegate to subordinate agencies, in individual cases. In general, independent experts or expert organisations, namely the Technical Inspection Agencies (TÜV) are involved.

2.1. The Atomic Energy Act

The Atomic Energy Act regulates in particular:

- import and export of nuclear fuel (Section 3 AtG);
- transportation of nuclear material (Section 4, 4a, 4b AtG);
- safe custody, possession and delivery of nuclear fuel (Section 5 AtG);
- storage of nuclear fuel outside state custody (Section 6 AtG);
- construction, operation, ownership and decommissioning of stationary nuclear facilities for the production, processing, fission or reconditioning of irradiated nuclear fuel (Section 7 AtG);
- treatment, processing or other use of nuclear fuel outside of facilities subject to licensing Section 9 AtG);
- the use of radioactive residual materials and the safe disposal of radioactive wastes (Section 9a AtG).

2.2. Ordinances and technical rules

The provisions of the Atomic Energy Act are supplemented or specified by further laws and regulations including:

- Precautionary Radiation Protection Act (Strahlenschutzvorsorgegesetz, StrVG);
- Federal Water Act (Wasserhaushaltsgesetz, WHG);
- Federal Immission Control Act (Bundesimmissionsschutzgesetz, BImSchG);
- Environmental Impact Assessment Act (Gesetz über die Umwelterträglichkeits-prüfung, UVPG).

The most important ordinances include:

- Radiological Protection Ordinance (Strahlenschutzverordnung, StrlSchV);
- Nuclear Licensing Procedures Ordinance (Atomrechtliche Verfahrens-verordnung, AtVfV);
- Financial Security Ordinance implementing the Atomic Energy Act (Atomrechtliche Deckungsvorsorge-Verordnung, AtDeckV);
- Atomic Law Cost Ordinance (Atomrechtliche Kostenverordnung, AtKostV);
- Nuclear Safety Officer and Reporting Ordinance (Atomrechtliche Sicherheitsbeauftragten- und Meldeverordnung, AtSMV);
- Final Storage Advance Payments Ordinance (Endlagervorausleistungs-verordnung, Endlager VlV);
- X Ray Ordinance (Röntgenverordnung, RöV).

The safety requirements have not been specified in great detail. Thus, there is room for different technical solutions, which, however all have meet the same protection goal. The licensing and supervisory authorities then have to examine whether this target is actually met. Licences for the construction and operation of nuclear facilities pursuant to section 7 AtG are granted by the supreme Länder authorities, generally the Ministries for the Environment of the Länder. These are also responsible for the supervision of the nuclear facilities in operation. The Länder Committee for Nuclear Energy and its technical committees, which are presided over by the BMU, serve an exchange of experience between the Federal Government and the Länder and co-ordination to ensure a uniform procedure of all Länder in the field of nuclear safety and radiation protection.

3. ENVIRONMENTAL STATUS OF FUEL FABRICATION FACILITIES

3.1. Effluents (radiological and non-radiological)

In Germany, the discharge of radioactive effluents in exhaust air and waste water from nuclear facilities is determined by the operators of the individual facilities and reported to the competent supervising authorities, in accordance with the "Guideline relating to Emission and Immission Monitoring of Nuclear Facilities". Details on the extent of measurements, measurement procedures, sampling, instrumentation and documentation of measuring results are stipulated in the rules of the Nuclear Safety Standards Commission (KTA). The measurements to be carried through by the operators of the facilities are checked by means of control measurements from experts called-in by the competent authorities, in accordance with the guideline relating to the "Control of Self-Surveillance of Radioactive Effluents from Nuclear Power Plants".

The values determined for the discharge of radioactive effluents in the years 1991 to 1997 show that the limits of emission per year, stipulated by the competent authorities, were observed in all cases. In general, the actual effluents per year have been far below the licensing values. Tables I and II include details on the discharge of radioactive effluents in exhaust air and waste water from nuclear fuel processing plants.

3.2. Environmental sampling and monitoring

In order to identify the distribution of released radioactive effluents into the environment, environmental monitoring programmes have been developed for facilities of the nuclear fuel cycle. These programmes include:

- routine soil sampling for uranium measurement; the samples are particularly taken in the area of the main distribution line, i.e. in a distance of up to 5 km;
- routine uranium measurement in surface waters.

The available incident precaution programmes (Störfallvorsorgeprogramme) regulate the measurements of radioactive effluents after their accidental release. By means of these programmes, immediate and reliable information on the potential distribution of released material is possible. It can be ensured with regular training that in the case of incident, the personnel is acquainted with the precautionary measurement programme.

A separate, from the plants independent measurement system for the Surveillance of the Environmental Radioactivity in Germany is the Integrated Measurement and Information System IMIS. IMIS is part of the German Government's National Response Plan for dealing with the consequences of a large scale contamination of the environment. It has two modes of operation, e.g., normal and emergency operation. During normal operation IMIS continuously monitors and documents the radiological background situation all over the country and provides prompt early warning in anomalous situations. In emergency situations IMIS provides all data and information needed for a fast and comprehensive assessment of the situation and the recommendation of measures for the protection of the population and the reduction of the contamination of the environment. IMIS is designed to operate at three levels, e.g., data collection, data assessment and decision making including countermeasures.

TABLE I. EMISSION OF RADIOACTIVE MATERIAL IN EXHAUST AIR FROM THE NUCLEAR FUEL CYCLE FACILITIES

(Bq ∝-activity0

	SIEMENS - U Lingen (ANF)	**SIEMENS - U Hanau**	**SIEMENS – MOX Hanau**	**SIEMENS - U Karlstein**	**NUKEM Hanau**
1994	< 1.8 E+04	< 1.4 E+06	< 1.4 E+04	< 2.4 E+03	4 E+05
1995	< 1.7 E+04	< 8.5 E+05	< 1.8 E+04	< 2.3 E+03	2.1E+05
1996	< 1.6 E+04	< 7.1 E+05	< 2.6 E+04	< 2.7 E+03	5.0 E+05
1997	< 1.6 E+04	< 2.6 E+05	< 2.9 E+04	< 1.6 E+03	3.1 E+05

TABLE II. EMISSION OF RADIOACTIVE MATERIAL IN WASTE WATER FROM NUCLEAR FUEL CYCLE FACILITIES

(Bq ∝-activity)

	SIEMENS - U Lingen (ANF)	**SIEMENS - U Hanau**	**SIEMENS – MOX Hanau**	**SIEMENS - U Karlstein**	**NUKEM Hanau**
1994	[a]	8.5 E+08	< 1.5E+05	1.5 E+07	6 E+07
1995	[a]	3.1 E+08	< 1.5E+05	[a]	1.9 E+08

[a] No data provided

At level 1 (data collection) 5 nation wide on-line networks, 44 specialised laboratories and about 30 mobile units are operated. The on-line networks continuously monitor the gamma dose rate at 1 m above ground, the activity concentration in air and the gamma dose rate in rivers and coastal waters. The network for gamma dose rate at 1 m above ground consists of 2 150 stations. In the emergency mode the on-line networks of IMIS provide synoptic data sets every 2 hours. The obligation of the 44 specialised laboratories is to collect and measure various types of environmental samples; the most important samples are milk, food, and drinking water which via ingestion may substantially contribute to the total dose. The application of standardised techniques for sample collection and measurement (mostly high resolution gamma spectrometry) is mandatory for all laboratories of IMIS. The mobile units have the obligation to provide data which „interpolate" between the results at fixed stations or to trace a plume in air or in water in a fast and reliable way. In situ gamma spectrometry is an important technique applied by the mobile units. All organisations and units with responsibilities for data collection regularly participate in quality assurance programmes.

At level 2 all data which are collected at level 1 are processed and circulated by a computer-based information system and are stored in a central data base. At this level dedicated,

computer-based model systems for the diagnosis and the prognosis of the long range atmospheric transport and of the radiological situation (dose and contamination) in Germany are operated.

The measured data and the model results provide the basis for the assessment of the contamination situation and for decision making by the Federal Ministry of Environment, Nature Conservation, and Nuclear Safety (level 3). The recommendation of protective measures for the population, the enforcement of food bans and the information of the public are further obligations at this level.

3.3. Radiation exposure estimation

Calculated values of radiation exposure in the vicinity of nuclear facilities are based on the results of emission monitoring. Radiation exposure of the general public in the environment of nuclear facilities is determined for the reference man, as is defined in the Radiation Protection Ordinance (Strahlenschutzverordnung, StrlSchV). This is in accordance with the procedure stipulated in the "General Administrative Provisions“ to Section 45 of the Radiation Protection Ordinance (StrlSchV). The exposure values indicated in the figure present upper values, which were determined at the most unfavourable point of impact for a reference man. For the calculation of these values, the exposure pathways defined in the Radiation Protection Ordinance and the habits of the reference man were considered.

For the nuclear fuel processing plants in Hanau, Karlstein and Lingen, Table III shows the upper values of effective dose for adults and infants as well as the upper values of lung dose for infants from emissions of radioactive effluents in exhaust air, calculated for the most unfavourable point of impact. The highest value of effective dose is 0.5 µSv for adults (approximately 0.2 % of the dose limit), the highest value of lung dose is 1.0 µSv for infants (approximately 0.2 % of the dose limit).

TABLE III. UPPER VALUES OF RADIATION EXPOSURE IN THE VICINITY OF NUCLEAR FACILITIES FROM EXHAUST AIR IN 1995 AND 1996

(µSv)

Owner	Plant/site	upper values					
		effective dose				lung dose	
		for adults		for infants		for infants	
		1995	1996	1995	1996	1995	1996
NUKEM GmbH	Hanau	0.2	0.5	0.2	0.5	0.5	1.0
SIEMENS AG	MOX-plant/Hanau	< 0.1	< 0.1	< 0.1	< 0.1	0.1	< 0.1
	U-plant/Hanau	< 0.1	< 0.1	< 0.1	< 0.1	< 0.3	< 0.2
	U-plant/Karlstein	< 0.1	< 0.1	< 0.1	< 0.1	< 0.1	< 0.1
	ANF GmbH/Lingen	< 0.1	< 0.1	< 0.1	< 0.1	< 0.1	< 0.1

3.4. Technologies for health and environment

The basic rules in the application of technologies for the protection of health and environment, in case of establishing nuclear fuel processing plants in Germany, are the corresponding Safety Criteria. The Safety Criteria are essentially a more detailed determination of the safety related “state of science and technology” as it is demanded in general by the Atomic Energy Act and the Radiation Protection Ordinance. The Safety Criteria were used in the licensing procedure of German fuel fabrication plants, namely the former RBU and ALKEM at Hanau and the ANF facility at Lingen, which are now all owned by Siemens AG.

In the following, the main emphasis of the safety criteria are listed for selected issues:

3.4.1. Radiation protection ordinance

The radiation protection is based on ICRP Principles. The basic requirements are in detail formulated in the "Ordinance about Protection against Injuries by Ionizing Radiation". Protection of employees as well as protection of population and environment have to be realised in a sufficient manner.

Limitations of individual doses for population are given for normal operation including deviations (para 44) and for design basis accidents (para 28.3). For the calculation of dose values for population the doses by direct radiation, inhalation and intake of radionuclides with food have to be taken into account. For design basis accidents the accumulated dose values up to 50 years must be calculated. Special guidelines how to perform these calculations are issued.

For workers in the plant, the given individual dose limitations are based on external radiation and incorporation. The main annual radiation dose limits for normal operation for workers and population are shown in Table IV. Additionally to these limits the 'ALARA' principle ("as low as reasonably achievable") has to be attended for design as well as in operation.

TABLE IV. MAIN GERMAN ANNUAL RADIATION DOSE LIMITS

(mSv/a)

	Employees	**Population**		
	18 or over	**Direct Radiation**	**Aerial Discharges**	**Liquid Discharges**
Whole body effective dose	50	1.5	0.3	0.3
Individual organs and tissues, lens of the eye	150	[a]	0.9	0.9
Thyroid	150	[a]	0.9	0.9
Bone Surface, Skin	300	[a]	1.8	1.8
Hands, Arms, Feet	500	[a]	[a]	[a]

[a] Not specified

Note: The German Annual Radiation Dose Limits will be adapted to the EU Basic Principles in the year 2000

3.4.2. Fire and explosion protection

Prevention of uncontrolled fire and explosion is an important safety goal especially in MOX fuel fabrication, but also in UO_2 fuel fabrication and enrichment. To ensure, that this goal will be reached, different types of safety measures are required.

Criteria for design and construction:

- Separation of fabrication and storage sections by fireproof walls;
- As far as possible use of fire resistant materials for construction of container and glove boxes;
- Separation of safety systems, ventilation equipment, off-gas filters;
- Separation of supply systems for hydrogen and other burnable media;
- Protection of ventilation systems.

Criteria for operation:

- Avoidance of ignition sources and if possible inflammable materials (for instance gas mixtures, surveillance of radiolysis gas production);
- Operation of fire detection systems.

Criteria for fire fighting:

- Stationary and mobile fire fighting systems;
- Automatic fire extinguishing systems for areas with dispersible Pu and Pu storage or with difficult access;
- Fire fighting by water only outside of areas with criticality risk
- Strict partitioning of the plant into fire zones, automatic separation of connections (pipes, ventilation system) in case of fire;
- automatic operation of extinguishing systems (sprinkler, gas), started by fire detectors;
- fire detectors in glove boxes;
- an own specially equipped fire brigade.

3.4.3. Criticality safety

Criticality safety in handling, manufacturing and storage of fissile material is based on the double failure principle, which is in detail written in the German standard DIN 25 403.

In general the proof of criticality safety is based on calculations. For systems with simple geometric shape (cylinder, slab or sphere) data from handbooks may be use, for instance the German criticality safety handbook (ed. GRS). Suitable safety margins must be taken into account. In the handbook the safety factors, which shall be applied to critical limits are given.

In more complicated cases like arrays of fissile material or special geometric shapes criticality safety has to be proved by calculation. Generally a value of $k_{eff} < 0.95$ is required including uncertainties. For normal operation, those conditions which lead to the highest k-value have to be considered, such as the highest density of fissile material, moderation, chemical compound etc. For example, in the criticality analysis of MOX fuel fabrication a residual

humidity of 3 wt % of water was assumed for dry MOX. Neutronic interaction of adjacent components must be checked, interspersed moderation effects are to be taken into account.

To realise criticality safety, essential safety concepts, which are described in detail in the DIN 25 403 standard are recommended:

- safe geometry, limitation of dimensions;
- mass limitation, double batching has to be considered;
- moderation control, to be applied only for handling of dry material;
- limitation of enrichment, control by at least two independent ways is required;
- if neutron poisoning is used as part of a safety concept, additional measures for quality assurance are required, to ensure the efficacy of the neutron absorber.

Priority should be given to technical safety measures, such as limitation of dimensions instead of administrative ones.

A criticality alarm system is required for all areas, were amounts of fissile materials are handled or stored, which are large enough to reach criticality under certain conditions. Instructions for the alarm case have to be provided.

3.5. Energy consumption of facility operation

For the SIEMENS fuel fabrication plant at the Lingen site, (see description Section 1.1) the following data for the energy consumption are available:

Electricity	2.5×10^4 kWh/tU
Fuel and Hydrogen	3.4×10^7 kJ/tU

The equivalent capacity is based on 400 t UO_2/y.

4. OPERATOR EXPOSURE

Average whole body exposure (mSv) and collective whole body exposure (man, mSv) for employees of the ANF fuel fabrication facility at Lingen are given in Table V. The data are taken from annual reports from the period 1991 to 1998.

The individual doses for employees of the ANF fuel facility at Lingen on exposure range (mSv) and number of workers in a given range over the same period are given in Table VI.

TABLE V. AVERAGE AND COLLECTIVE WHOLE BODY EXPOSURE FOR EMPLOYEES OF THE ANF FUEL FABRICATION FACILITY AT LINGEN

	Average whole body exposure (mSv)	**Collective whole body exposure** (man mSv)
1991	0.52	116.2
1992	0.38	101.4
1993	0.63	178.6
1994	0.69	197.0
1995	0.41	120.8
1996	0.46	145.7
1997	0.60	180.9
1998	0.77	228.0

TABLE VI. INDIVIDUAL DOSES FOR EMPLOYEES OF THE ANF FUEL FABRICATION FACILITY AT LINGEN

Exposure range	**Number of workers in given range**							
(mSv)	**1991**	**1992**	**1993**	**1994**	**1995**	**1996**	**1997**	**1998**
0,2 - < 2	108	107	197	120	124	171	161	182
2 - < 5	10	17	25	20	5	18	25	34
5 - < 10	2	1	2	3	2	0	0	0
10 - < 15	0	1	0	0	0	0	0	0
> 15	0	0	0	0	0	0	0	0
Total	120	126	224	143	131	143	186	216

JAPAN

1. STATUS OF FUEL FABRICATION FACILITIES

In Japan, there are five fuel fabrication facilities owned by four enterprises which manufacture uranium fuel assemblies for light water reactors and three MOX fuel fabrication facilities operated by JNC (Japan Nuclear Cycle Development Institute, former PNC). Table I gives an overview of the facilities and shows the maximum production capacities.

At present, JNC owns and operates three MOX fuel facilities at Tokai Works. These facilities, which are called respectively Plutonium Fuel Development Facility (PFDF), Plutonium Fuel Fabrication Facility (PFFF) and Plutonium Fuel Production Facility (PFPF), have the purpose to develop the fuel fabrication technology and to fabricate MOX fuel assemblies.

At the PFDF, installed in 1966, JNC fabricates special test fuel and Pu thermal test fuel, which are used for irradiation tests respectively by the experimental fast reactor JOYO (100MWt) and by light water reactors. The facility also carries out research and development on the MOX fuel technology.

The PFFF has been operated since 1972 and has fabricated MOX fuel for the prototype advanced thermal reactor (ATR) FUGEN (165MWe) which was uniquely developed by JNC. The present production capacity of the PFFF is 10 ton MOX per year.

The PFPF started its operation in 1988 and has fabricated MOX fuel assemblies for the prototype fast breeder reactor MONJU (280 MWe) and the experimental fast reactor JOYO. The present production capacity of the PFPF is 5 ton MOX per year. The cumulative MOX fuel production of PFDF, PFFF and PFPF from 1969 to 1998 is about 156 ton.

2. STATUS OF THE SAFETY REGULATION PROCEDURE FOR NUCLEAR FUEL FACILITIES

2.1. Outline

In order to license nuclear fuel facilities to operate, each separately defined stage is examined in accordance with the *Law for the Regulation of Nuclear Source Material, Nuclear Fuel Material and Reactors*. There are six separate stages: “before construction”, “ under construction”, “before operation”, “ adequacy of the design”, “ procedure of construction” and “ operational procedure and maintenance”.

In the Law, the nuclear fuel facilities are divided into the categories fabrication facility, reprocessing facility and nuclear fuel material user. The regulation requirements are different for each category. The uranium fuel fabrication facility is prescribed in the category of fabrication facility and the MOX fuel fabrication facility is prescribed in the category of nuclear fuel material user.

TABLE I. NUCLEAR FUEL FABRICATION FACILITIES IN JAPAN
(As of March 31, 1999)

Undertaker	Location	Enrichment %	Capacity t U/a	Remarks
Japan Nuclear Fuel Conversion(JCO)	Tokai-mura Naka-gun Ibaraki pref.	≤5	715	Conversion for BWR and PWR
		≤20	3	Conversion for Research Reactor
Mitsubishi Nuclear Fuel Co., Ltd.(MNF)	Tokai-mura Naka-gun Ibaraki pref.	≤5	475	Conversion for PWR
		≤5	440	Fuel rod for PWR
Nuclear Fuel Industries, Ltd. Tokai Works (NFI Tokai)	Tokai-mura Naka-gun Ibaraki pref.	≤5	200	Fuel rod for BWR
Nuclear Fuel Industries, Ltd. Kumatori Works (NFI Kumatori)	Kumatori-machi Sennan-gun Osaka	≤5	284	Fuel rod for PWR
		≈90	0.475	Fuel plate for Research Reactor
Japan Nuclear Fuel Co., Ltd. (JNF)	Yokosuka City Kanagawa pref.	≤5	750	Fuel rod for BWR
Japan Nuclear Cycle Development Institute Tokai Works (JNC Tokai)	Tokai-mura Naka-gun Ibaraki pref.	≤2.7[a]	10[b]	Fuel rod for ATR (PFFF)
		≤32[c]	5[b]	Fuel rod for FBR (PFPF)

[a] $(^{239}Pu + ^{241}Pu + ^{235}U)/(Pu+U)$ [b] t MOX/a [c] Pu/(Pu+U)

2.2. Procedure of safety regulation

2.2.1. Fabrication facility

Any person, who wishes to commence a fabrication business, shall obtain permission from the Minister of Economics, Trade and Industries. Prior to the permission, a review shall be made of the fabrication capability, the technical capability, the financial basis and the prevention of any hazard, taking into consideration the Safety Review Guideline issued by the Nuclear Safety Commission (NSC). Before licensing, the Minister of Economics, Trade and Industries shall inquire the opinions of Atomic Energy Commission (AEC) and NSC with respect to the application of standards for the permission.

Following the permission, the Minister of Economics, Trade and Industries gives to the applicant the authorization of the "design" and the "construction procedure" before construction start. Subsequent procedures are the inspection by the Minister of Economics, Trade and Industries before commencing operation of the facility and finally to acquire his authorization of the safety regulation for the plant operation, which must proceed the start of the business.

As for welding in the fabrication facilities, any fabrication enterprise shall obtain the authorization of the Minister of Economics, Trade and Industries with respect to welding methods and prior to his inspection.

After starting operation, any fabrication enterprise shall be obligated to observe the safety regulations, to keep operational records and to report failures and incidents occurred in the relevant facilities.

2.2.2. Nuclear fuel material user

PFDF, PFFF and PFPF operated by JNC are employed for research and development on MOX fuel fabrication technology but not for commercial fuel fabrication. In the Law, these facilities are classified as nuclear fuel material user and the procedure of safety regulation is different from that for commercial fabrication facilities. The authorization of the "design' and the "construction procedure" by the Minister of Education, Culture, Sports, Science and Technology is not necessary for these facilities and inquiries from AEC and to NSC are not needed for the nuclear fuel material user. After the review of the application documents by the Minister of Education, Culture, Sports, Science and Technology will give the permission, consulting to the Safety Review Guideline.

2.3. Radiation exposure and annual limit on intake

The annual limit of radiation exposure for employees and the public, the annual limit on intake for employees, the derived air concentration for employees and the derived concentration for the public are given in the Law (see Tables II and III). These values are based on ICRP [27, 28].

3. ENVIRONMENTAL STATUS FUEL FABRICATION FACILITIES

3.1. Effluent (radiological and non-radiological)

3.1.1. Radioactive gaseous waste

The working areas in uranium fuel fabrication facilities are divided into areas where radioactive contamination is possible and where it is not. In the former areas, uranium is handled in the sealed equipment or in the hoods, which are maintained with a negative air pressure to prevent contamination of the working areas. Air in the hood is filtered through double cascaded, high efficiency particulate air filters and then is discharged through a stack or exhaust outlet. Air in the working area is also purified with such filters and discharged through the same stack or exhaust outlet. The effluent to be released into the atmosphere is continuously monitored for alpha radioactivity in the exhaust particulate.

In the MOX fuel fabrication facilities, non-sealed nuclear fuel material is only handled in airtight glove boxes. During normal operation of the facility, the working environments are, therefore, free from radioactive contamination. The air in the glove boxes, in which nuclear materials are handled in the non-sealed condition, is filtered by the high efficiency particulate air filter and mixed with the air ventilated from the working environment to be released through the stack into the atmosphere. The alpha activity in the air to be discharged in the atmosphere is continuously monitored with exhaust monitoring device. Table IV shows the emission of gaseous waste over the period 1994 to 1998.

TABLE II. ANNUAL RADIATION DOSE LIMITS

(mSv/a)

	Occupational	**Public**
Effective Dose Equivalent	50	1
The lens of the eye	150	50
Individual organs except the lens of the eye	500	50 (for skin)

Note: For woman is abdomen; 13mSv/3month (only occupational, possible to pregnancy); 10mSv (only occupational, until the birth since pregnancy)

TABLE III. ANNUAL LIMIT ON INTAKE AND DERIVED LIMIT FOR URANIUM, PLUTONIUM AND AMERICIUM

Nuclide	**Chemical form**	**Annual limit intake** Bq		**Derived limit** Bq/cm^3		
		inhalation	ingestion	DAC[a]	Air[b]	Water[c]
^{234}U	UF_6	$6.9\cdot10^4$ ($4.5\cdot10^4$)	$7.0\cdot10^5$ ($4.5\cdot10^5$)	$2\cdot10^{-5}$	$2\cdot10^{-7}$	$1\cdot10^{-2}$
	UO_2, U_3O_8	$1.3\cdot10^2$	$7.3\cdot10^6$	$5\cdot10^{-7}$	$3\cdot10^{-9}$	$2\cdot10^{-1}$
^{235}U	UF_6	$7.5\cdot10^4$ ($5.0\cdot10^4$)		$2\cdot10^{-5}$	$2\cdot10^{-7}$	$2\cdot10^{-2}$
	UO_2, U_3O_8	$1.5\cdot10^3$	$7.3\cdot10^6$	$5\cdot10^{-7}$	$4\cdot10^{-9}$	$2\cdot10^{-1}$
^{238}U	UF_6	$7.7\cdot10^4$ ($5.1\cdot10^4$)		$2\cdot10^{-5}$	$2\cdot10^{-7}$	$2\cdot10^{-2}$
	UO_2, U_3O_8	$1.5\cdot10^3$	$8.1\cdot10^6$	$5\cdot10^{-7}$	$4\cdot10^{-9}$	$2\cdot10^{-1}$
^{238}Pu	oxide	$6.5\cdot10^2$	$3.7\cdot10^6$ ($3.1\cdot10^6$)	$2\cdot10^{-7}$	$2\cdot10^{-9}$	$8\cdot10^{-2}$
^{239}Pu	oxide	$6.1\cdot10^2$ ($5.9\cdot10^2$)	$3.5\cdot10^6$ ($2.7\cdot10^6$)	$2\cdot10^{-7}$	$1\cdot10^{-9}$	$7\cdot10^{-2}$
^{240}Pu	oxide	$6.1\cdot10^2$ ($5.9\cdot10^2$)	$3.5\cdot10^6$ ($2.7\cdot10^6$)	$2\cdot10^{-7}$	$1\cdot10^{-9}$	$7\cdot10^{-2}$
^{241}Pu	oxide	$3.7\cdot10^4$ ($2.7\cdot10^4$)	$2.4\cdot10^8$ ($1.4\cdot10^8$)	$9\cdot10^{-6}$	$9\cdot10^{-8}$	$5\cdot10^0$
^{242}Pu	oxide	$6.4\cdot10^2$ ($6.3\cdot10^2$)	$3.7\cdot10^6$ ($2.9\cdot10^6$)	$2\cdot10^{-7}$	$2\cdot10^{-9}$	$8\cdot10^{-2}$
^{241}Am	all forms	$4.1\cdot10^2$ ($2.2\cdot10^2$)	$5.0\cdot10^4$ ($2.7\cdot10^4$)	$8\cdot10^{-8}$	$1\cdot10^{-9}$	$1\cdot10^{-3}$

[a] derived air concentration (DAC) (for occupational)

[b] concentration limit for air at the outside of supervised area (for public)

[c] concentration limit for water at the outside of supervised area (for public)

note: values in brackets show the value for the bone surface

TABLE IV. QUANTITIES OF GASEOUS WASTE DISCHARGE

GBq

Enterprise	1994/1995	1995/1996	1996/1997	1997/1998	1998/1999
JCO	$< 4.3 \cdot 10^{-4}$	$< 4.3 \cdot 10^{-4}$	$< 5.3 \cdot 10^{-4}$	$< 4.3 \cdot 10^{-4}$	$< 4.1 \cdot 10^{-4}$
MNF	$< 3.5 \cdot 10^{-4}$	$< 3.5 \cdot 10^{-4}$	$< 3.4 \cdot 10^{-4}$	$< 3.4 \cdot 10^{-4}$	$< 3.9 \cdot 10^{-4}$
NFI Tokai	$< 2.1 \cdot 10^{-4}$	$< 2.1 \cdot 10^{-4}$	$< 2.9 \cdot 10^{-4}$	$< 2.1 \cdot 10^{-4}$	$< 1.9 \cdot 10^{-4}$
NFI Kumatori	$< 2.0 \cdot 10^{-4}$	$< 2.0 \cdot 10^{-4}$	$< 1.0 \cdot 10^{-4}$	$< 1.0 \cdot 10^{-4}$	$< 2.0 \cdot 10^{-4}$
JNF	$< 0.9 \cdot 10^{-4}$	$< 0.9 \cdot 10^{-4}$	$< 0.8 \cdot 10^{-4}$	$< 0.8 \cdot 10^{-4}$	$< 0.8 \cdot 10^{-4}$
JNC Tokai	$< 7.1 \cdot 10^{-4}$	$< 7.1 \cdot 10^{-4}$	$< 7.1 \cdot 10^{-4}$	$< 7.1 \cdot 10^{-4}$	$< 7.1 \cdot 10^{-4}$

note: fiscal years from April to March

3.1.2. Radioactive liquid waste

At a uranium fuel fabrication facility, waste water is treated by an apparatus to remove radioactive materials. Waste water to be discharged is collected in a storage tank and monitored for the concentrations of radioactivity before release to the sea or into the river. Release will be made after the reconfirmation that the radioactive concentration is under the reference value.

Liquid waste which arises from the controlled area in the MOX facilities include water for hand-washing of workers, laundry waste water of working clothes and decontaminated water from the decontamination room. There is no possibility for hand-washing water to be contaminated and during normal facility operation, there is little possibility of contamination of the laundry water, because of the strict contamination survey of personnel after work. Except in case of an accident, contaminated water will not be generated in the decontamination room. The radioactive liquid waste is collected in the storage tank and batch wise released through the ocean discharge pipe, if the radioactive concentration is under the reference value. Table V shows the emission of liquid waste over the period 1994 to 1998.

TABLE V. QUANTITIES OF LIQUID WASTE DISCHARGE

GBq

Enterprise		1994/1995	1995/1996	1996/1997	1997/1998	1998/1999
JCO		$< 3.0 \cdot 10^{-2}$	$< 3.1 \cdot 10^{-2}$	$< 3.5 \cdot 10^{-2}$	$< 3.3 \cdot 10^{-2}$	$< 3.0 \cdot 10^{-2}$
MNF		$< 2.2 \cdot 10^{-2}$	$< 2.1 \cdot 10^{-2}$	$< 2.2 \cdot 10^{-2}$	$< 1.9 \cdot 10^{-2}$	$< 2.1 \cdot 10^{-2}$
NFI Tokai		$< 3.3 \cdot 10^{-4}$	$< 3.1 \cdot 10^{-3}$	$< 3.9 \cdot 10^{-3}$	$< 4.1 \cdot 10^{-3}$	$< 4.0 \cdot 10^{-3}$
NFI Kumatori		$< 2.0 \cdot 10^{-3}$	$< 2.0 \cdot 10^{-3}$	$< 2.0 \cdot 10^{-3}$	$< 2.0 \cdot 10^{-3}$	$< 2.0 \cdot 10^{-3}$
JNF		$< 9.1 \cdot 10^{-4}$	$< 7.6 \cdot 10^{-4}$	$< 7.1 \cdot 10^{-4}$	$< 6.6 \cdot 10^{-4}$	$< 6.6 \cdot 10^{-4}$
JNC Tokai	Pu	$< 4.1 \cdot 10^{-5}$	$\sim 1.8 \cdot 10^{-5}$	$\sim 4.6 \cdot 10^{-5}$	$\sim 5.0 \cdot 10^{-5}$	$\sim 1.2 \cdot 10^{-4}$
	U	$< 4.8 \cdot 10^{-4}$	$< 4.8 \cdot 10^{-4}$	$< 4.8 \cdot 10^{-4}$	$< 4.8 \cdot 10^{-4}$	$< 4.8 \cdot 10^{-4}$

note: fiscal years from April to March

3.1.3. Non-radioactive waste

From the viewpoint of environmental prevention, measurements of the general pollutants in the liquid waste are carried out together with the radioactive materials every time before emission. Measurement are performed on pH, COD, etc., based on the pollution control

regulation of the local government. The emission of liquid waste is allowed if the concentration of the radioactive material and the general pollutants are under the reference values. Table VI shows the reference values of each facility.

TABLE VI. REFERENCE LEVEL OF NON-RADIOACTIVE SUBSTANCES

Substances	**Unit**	**Reference level**		
		JNC	**MNF, JCO**	**JNF**
pH		5.8 to 8.6	5.0 to 9.0	5.8 to 8.6
COD	mg/l	<10	<160	<60
SS	mg/l	<20	<200	<90
Fluorine	mg/l	<8	<15	<15
Normal hexane	mg/l	<5	-[a]	<2
Organophosphorus	mg/l	<1	-[a]	<1
Nitrogen compound	mg/l	-[a]	-[a]	<60

note: the pollution control regulation of the local government does not apply to NFI-Tokai and NFI-Kumatori; [a] not specified

3.2. Environmental sampling and monitoring

It is prescribed in the Law (the Law for the Regulation of Nuclear Source Material, Nuclear Fuel Material and Reactors) that the effective dose equivalent for the public caused by the radioactive effluent released from uranium fabrication facility and MOX facility should not exceed 1 mSv per year.

As shown in Tables IV and V, the quantity of radioactive waste discharge into the atmosphere and the open sea is so small that the concentrations of gaseous and liquid waste will be well below the derived concentration for the public at the release point of the facility.

The radioactive effluent released from the facilities will be diluted further by the atmospheric and the oceanic movement respectively for the gaseous and the liquid effluents so that the dose equivalent for the public at the site boundary will be significantly below the public dose limit 1 mSv.

It can be concluded from the above circumstances, that during normal operation of the facility there is no need of a specific environmental monitoring programme for the uranium fabrication facility and the MOX fuel fabrication facility. However JNC practices the environmental monitoring in and around the Tokai-Works, where also a reprocessing facility is installed and operated. The information resulting from the measurement of radioactivity in the environment are published to demonstrate to the public that the safety of the surrounding environment is well maintained.

3.3. Estimation of radiation exposure for the public

At each nuclear fuel fabrication facility, the enterprise is obliged to keep the radioactive concentration of gaseous and liquid waste under the concentration limits which are prescribed

in the Law. Reason that for the Japanese nuclear fuel fabrication facilities, the evaluation of radiation exposure for the critical group at the stage of safety examination for permission is not necessary. However, in order to certify that the radiation exposure for the public satisfies the ALARA principle, the radioactive exposure for the public at the boundary of the supervised area of the site is evaluated at the stage of the safety examination for permission. The uranium isotopes and radioactive impurities such as ^{99}Tc, ^{237}Np, ^{106}Ru, ^{125}Sb, Pu(α) and Pu(β) are considered to be a radioactive source in the discharge.

The estimate values of the effective public dose at the boundary of the supervised area of the site are presented in Table VII. The actual values would be much smaller than the listed values which are evaluated with conservative assumptions. The dose estimation for the public based on the measured quantity of the radioactive material release through the facility operation is not carried out, because of the small concentration of effluent which is significantly under the derived concentration for the public.

TABLE VII. ESTIMATED EFFECTIVE DOSE EQUIVALENT FOR THE PUBLIC

(mSv/year)

Enterprises		**Direct and Skyshine**	**Gaseous Waste**	**Liquid Waste**
JCO		$4.0\cdot10^{-2}$	$3.0\cdot10^{-3}$	$6.0\cdot10^{-5}$
MNF		$3.0\cdot10^{-2}$	$9.0\cdot10^{-3}$	$1.0\cdot10^{-4}$
NFI Tokai		$4.9\cdot10^{-2}$	$3.4\cdot10^{-3}$	$2.2\cdot10^{-5}$
NFI Kumatori		$3.3\cdot10^{-2}$	$3.0\cdot10^{-3}$	$8.0\cdot10^{-3}$
JNF		$5.0\cdot10^{-2}$	$3.0\cdot10^{-3}$	$1.0\cdot10^{-3}$
JNC	PFFF	$8.4\cdot10^{-2}$	$<6.3\cdot10^{-5}$	$<1.4\cdot10^{-3}$
	PFPF	$1.1\cdot10^{-3}$	$<3.8\cdot10^{-6}$	[a]

[a] not specified

3.4. Technologies for health and environment

An advanced transparent radiation shielding material against fast neutrons was developed by JNC and the shielding producer. The shielding material was manufactured by using cyclo-olefin co-polymers as raw material. The properties of this shielding material are shown in the Table VIII.

TABLE VIII. THE PROPERTIES OF THE SHIELDING MATERIAL

		Acrylic resin	**Cyclo-olefin co-polymers**
Elemental constitution	wt%	Hydrogen : 8% Carbon : 60% Oxygen : 32%	Hydrogen : 12.00% Carbon : 87.77%
Density	g/cm^3	1.19	1.02
Density of hydrogen	g/cm^3	0.095	0.122
Total light transmittance	%	94.1	90.0
Flexural strength	MPa	90.2	83
Deflection temperature	^oC	100	77

The shielding property against fast neutrons was evaluated by experiments using ^{252}Cf and by calculation using MOX fuel as a neutron source. As a result, it was clarified that this shielding material had better shielding properties than that of acrylic resin, which is a typical transparent shielding material with similar optical and mechanical properties as those of acrylic resin. This shielding material was adopted as panels for some of the glove boxes in the MOX fuel fabrication facility of JNC. A reduction of the operator exposure is expected.

3.5. Energy consumption

The energy consumption of the different fuel facilities is given in Table IX.

TABLE IX. ENERGY CONSUMPTION OF NUCLEAR FUEL FABRICATION FACILITIES

(kWh/kg U)

Enterprises	1995	1996	1997	1998
JCO	15	16	16	17
MNF	58	57	60	58
NFI-T	55	47	56	59
NFI-K	70	75	65	60
JNF	37	41	40	47
JNC*	3370	3370	3370	3370

*kWh/kgMOX

4. OPERATOR EXPOSURE

At the uranium fuel fabrication facilities, both the specific radioactivity of uranium and the gamma dose rate in the working environment are low. This means that neither special gamma shielding for the facilities, nor strict containment is necessary, in contrast to the MOX facility. Workers at uranium fabrication facilities, therefore, receive relatively small radiation exposure. In the MOX fuel fabrication facility, as a result of the radiological characteristic of plutonium, some careful considerations for the operator exposure, in addition to those for the uranium fuel fabrication facility must be taken into consideration. The trends of radiation exposure for the employees are presented in Table X.

As plutonium has a larger specific activity and is more toxic than uranium, MOX fuel must be handled in perfectly sealed glove boxes. This leads to the more constraining conditions of the facility operability and also requires longer time for periodic inspection and maintenance of equipment. On the other hand, the spontaneous fission of plutonium and the alpha-neutron reaction in the working environment give rise to a higher neutron dose equivalent. The buildup of ^{241}Am in the fabrication process, which is the decay product of ^{241}Pu, increases also the gamma exposure in the working environment of the plant.

In the future, the increase of the exposure dose in the working environment can be one of the major concerns due to the growing amount of recycled plutonium containing higher numbers of plutonium isotopes. In order to maintain the level of radiation protection for the plant operator well below the individual dose limit of the ICRP, the future MOX fuel fabrication facility will have to adopt a fuel fabrication technology based on a simply designed, remotely manipulated and automatically controlled process.

TABLE X. RADIATION EXPOSURE DOSES TO WORKERS

Period[a]	Dose Distribution (in number of persons)					Collective Dose	Mean Exp. Dose
	≤5 mSv	5–15 mSv	15–25 mSv	25–50 mSv	≥50 mSv	(man mSv)	(mSv)
JCO							
1994/1995	151	0	0	0	0	16	0.1
1995/1996	139	0	0	0	0	11	0.1
1996/1997	137	0	0	0	0	17	0.1
1997/1998	114	0	0	0	0	15	0.1
1998/1999	104	0	0	0	0	11	0.1
MNF							
1994/1995	432	0	0	0	0	148	0.3
1995/1996	426	0	0	0	0	158	0.4
1996/1997	406	0	0	0	0	102	0.3
1997/1998	416	0	0	0	0	100	0.2
1998/1999	381	0	0	0	0	95	0.2
NFI Tokai							
1994/1995	305	0	0	0	0	60	0.2
1995/1996	270	0	0	0	0	50	0.2
1996/1997	290	0	0	0	0	44	0.2
1997/1998	280	0	0	0	0	56	0.2
1998/1999	391	0	0	0	0	38	0.1
NFI Kumatori							
1994/1995	384	0	0	0	0	31	0.1
1995/1996	337	0	0	0	0	36	0.1
1996/1997	313	0	0	0	0	19	0.1
1997/1998	277	0	0	0	0	23	0.1
1998/1999	290	0	0	0	0	20	0.1
JNF							
1994/1995	711	0	0	0	0	197	0.3
1995/1996	718	0	0	0	0	167	0.2
1996/1997	739	0	0	0	0	84	0.1
1997/1998	601	0	0	0	0	56	0.1
1998/1999	651	0	0	0	0	75	0.1
JNC Tokai							
1994/1995	2403	45	0	0	0	860	0.4
1995/1996	3020	61	0	0	0	1002	0.3
1996/1997	2815	23	0	0	0	781	0.3
1997/1998	2968	0	0	0	0	235	0.1
1998/1999	3003	2	0	0	0	288	0.1

[a] fiscal years from April to March

SWEDEN

1. PRESENT STATUS OF FUEL FABRICATION FACILITIES

1.1. Westinghouse

The nuclear fuel division is one of three divisions within Westinghouse. The others are BWR Services and Nuclear Automation. In addition, there are two support functions. The fuel is manufactured at the factory in Västerås, Sweden.

The plant commenced its operation in 1971 and the products were BWR fuel elements and control rods. Today, the products are PWR and BWR fuel elements, BWR control rods, BWR fuel boxes and engineering services. The Westinghouse personal staff consists of approximately 830 persons, whereof 400 persons are employed at the fuel factory.

Approximately 50 % of the manufactured fuel is exported while the same figure for the control rods is 60–70%. The overall turnover per year is 900 million Swedish crowns. The capacity per year is 400 tons of uranium dioxide (UO_2). However, according to the concession from 1996 Westinghouse is allowed to produce 600 tons of UO_2 powder. The factory is licensed for a maximum enrichment of 5% ^{235}U. The current annual production capacities are 400 tons of UO_2 for BWR and PWR fuel, 300 BWR control rods and 1 800 BWR fuel boxes. Some interesting figures relating to the production are given in Table I.

TABLE I. PRODUCTION FIGURES

Type	Since	Amount
UO_2	1976	9 190 tons
BWR fuel elements	1976	26 926
PWR fuel elements	1983	1 098
Control rods	1970	4 154
Fuel boxes	1970	33 786

To produce the UO_2 powder, necessary for the consecutive pelletising process, the uranium hex fluoride has to be converted in the first step. Westinghouse utilises the process of wet conversion, developed by Siemens, to produce UO_2 powder. Westinghouse is the only factory in the world still using that method.

The Westinghouse nuclear fuel factory is certified according to ISO 9001, ISO 14001, USNRC 10 CFR 50 Appendix B, IAEA Safety Guide 50-C-QA, ANSI ASME NQA-1 and KTA 1401.

2. REGULATION PROCEDURE AND STATUS

The law within the nuclear field is stated in the form of acts and ordinances. The acts are published and written by the Swedish parliament and the ordinances by Swedish Radiation Protection Institute (SSI) and Swedish Nuclear Power Inspectorate (SKI). The environmental field is covered by the Swedish Environmental Authority. The most important acts and ordinances with relevancy to Westinghouse Atom activities are *summarised* (translation) below. Since most of the acts and ordinances are only written in Swedish, the translation is not official.

2.1. The act on nuclear activities (SFS 1984:3)

The Swedish government provided permission to Westinghouse according 5§ in the act on nuclear activities, on the 31st of May 1990.

1§ - This law applies to nuclear activities. Those activities are:

1. construction, possession or operation of nuclear constructions;
2. acquisition, possession, transfer, handling, manufacture, transport or other connections to nuclear substances or nuclear wastes;
3. imports of nuclear substances or wastes, to the kingdom of Sweden;
4. exports of nuclear wastes from the kingdom of Sweden.

5§ - Nuclear activities require permission in accordance with this act. Issues regarding permissions are tried by the government or competent authority decided by the government. Permit holder are allowed to assign someone else to take measures, that according to the act shall be done by the permit holder, after acknowledgement from the government or authority decided by the government.

2.2. Ordinance on nuclear activities (SFS 1984:14)

21§ - States that nuclear constructions, arrangements for possession, handling, manufacturing or transport of nuclear substances or nuclear wastes shall be tried, controlled or inspected to the extent that is needed to fulfil the safety requirements according to the act on nuclear activities (SFS 1984:3). The Swedish Nuclear Power Inspectorate informs about ordinances needed for that type of testing, control or inspection.

2.3. The Radiation Protection Act (SFS 1988:220)

6§ - Legal person who runs an activity with radiation shall with respect to the type of activity and the circumstances under which the activity is performed:

1. apply those measures and observe the precautionary measures needed to prevent damage to humans, animals or the environment;
2. control and establish radiation protection at the local place where radiation is present;
3. make sure that maintenance is performed on technical equipment, measure- and radiation protection equipment that are used in the activity.

13§ - Legal person who runs activity with radiation is responsible for the handling of radioactive wastes produced and, when is needed, that it is deposited finally in a safe manor from a radiation protection aspect. The same applies to discarded radiation sources used in the activity.

2.4. Nature Conservation Act (SFS 1964:822)

23§ - Every individual shall ensure not to litter plates, plastics, paper, wastes or others in nature or within occupied areas.

2.5. Environment Protection Act (SFS 1969:387)

1§ - This act applies to:

1. emissions of waste water, solid substance or gas from the ground, buildings, etc.
2. usage of the ground, building or construction in a manor that can cause pollution of the ground, the ground water, etc.
3. usage of the ground, building or construction in a manor that can disturb the surroundings by air pollution, noise, light or others if the disturbance is not occasional.

3. ENVIRONMENTAL STATUS OF THE WESTINGHOUSE NUCLEAR FUEL FACTORY

3.1. Discharges (radiological and non-radiological)

In accordance to Swedish law an operator of a nuclear facility has to report the amount of radioactive and non-radioactive discharges to the relevant authority. The Swedish Radiation Protection Institute (SSI) and Swedish Environmental Protection Agency (NVV) publish the necessary ordinances for that purpose. Once every third month the radiological report is written and sent to SSI. The report presents the radiological work along with the data concerning the environmental radioactive discharges. Corresponding report to NVV is compiled once per year. The radiological discharges are described as percentage of the reference discharge limits. The limits are 23 GBq and 1 TBq for air and water discharges respectively. The radiological effluents are described as percentage of the reference discharge limits. The limits are 23 GBq and 1 TBq for air and water discharges respectively.

3.1.1. Radiological discharges

Discharges to air and water from 1989 to 1996 are presented in Tables II and III The discharges are well below the reference discharge limit (see Figure 1). The competent authority (SSI) specifies that the nuclear installations have to comply with.

TABLE II. RADIOACTIVE AIR DISCHARGES

Reference discharge limit: 23 GBq

Year	Bq	% of limit
89	6.90E+06	0.030
90	8.05E+06	0.035
91	9.20E+06	0.040
92	6.90E+06	0.030
93	4.60E+06	0.020
94	3.34E+06	0.015
95	2.25E+06	0.010
96	2.05E+06	0.009
97	3.45E+06	0.015
98	4.14E+06	0.018
99	2.53E+06	0.011
00	9.20E+05	0.004

TABLE III. RADIOACTIVE WATER DISCHARGES

Reference discharge limit: 1 TBq

Year	Bq	% of limit
89	1.00E+08	0.010
90	1.00E+08	0.010
91	1.00E+08	0.010
92	2.00E+08	0.020
93	5.42E+08	0.054
94	8.63E+08	0.086
95	4.30E+08	0.043
96	4.41E+08	0.044
97	3.20E+08	0.030
98	3.30E+08	0.033
99	2.30E+08	0.023
00	1.60E+08	0.016

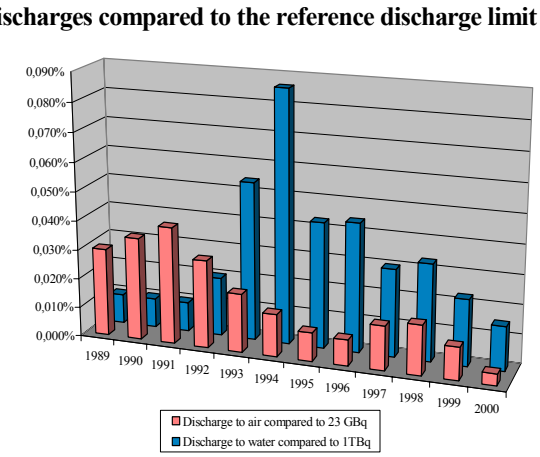

FIG. 1. The radiological air and water discharges respectively from the Westinghouse fuel factory in Västerås, Sweden.

From the beginning of the year 2002 the radiological discharges to the environment will be reported as absolute amount of activity. The Reference discharge limit will no longer be a valid measure of the releases. This fact is a consequence of the EU-directives that have come into effect since Sweden joined the European Union.

3.1.2. Non Radiological discharges

The non-radiological discharges are specified in Table IV and V for air and water respectively. The measurement results are reported to the County Administration.

TABLE IV**.** NON RADIOACTIVE AIRBORNE DISCHARGES
Total amount during 2000

Parameter	**Unit**	**Total amount**	**Limit 1996**	**Limit 2000**
U	kg	0.011	3	2
NO_x	kg	1 817	4 000	4 000
NH_3	kg	3 925	90 000	25 000
Methanol	kg	4 082	120 000	35 000
Ethanol	kg	310	150	1 000
VOC[a]	kg	4 392	[b]	[b]

[a] The total amount of discharges of volatile organic compounds to the air has been calculated and includes the discharges of methanol and ethanol. The used amount of volatile organic compounds has been calculated and the sum was reduced with the amount of hazardous waste.
[b] Not specified

3.2. Environmental monitoring and sampling

The radiological environmental monitoring programme for Westinghouse Atom is developed in co-operation with the Swedish Radiation Protection Institute. The programme includes radiological sampling of the uranium concentration in soil, rainwater, sludge and grass. The samples are taken at specific locations around the fuel factory. Four samples are located within the distance of 700 meters from the factory while the fifth is located 4 kilometres away from the factory at the community purification plant. The samples are analyzed with respect to the following isotopes ^{234}U, ^{235}U and ^{238}U. The results are reported to SSI once every 6 months.

TABLE V. NON RADIOACTIVE WATER DISCHARGES
TO THE PUBLIC SEWAGE SYSTEM

Total amount during 2000

Parameter	**Unit**	**System 222-3* Mini lime tower**	**System 763* Neutralisation**	**Total (excluding systems 222-3 and 763)**	**Limit (Year 1996)**	**Limit (Year 2000)**
Volume	m^3	1945	1248	1894		
pH		6.7	9.0			
U	kg	0.3	0.12	1.3	15	10
F	kg	23.7	5.8	7.6	225	150
NO_3	kg	32043	34443	21	195 000	2000
NH_3	kg	10313	0.00	22	15 000	500
Zr	kg		0.16			
Ni	kg		0.01			
Cr-t	kg		0.04			
Sn	kg		0.01			
Cu	kg		0.02			
Zn	kg		0.01			
Cd	kg		0.0003			
Pb	kg		0.0003			
Mo	kg		0.025			
Hg	kg		0.000			
P_{tot}	kg					

* Rest products that are directed to the municipal water purification plant by means of separate pipelines.
Empty fields means that no data were provided or that no limits are specified.

3.3. Radiation exposure estimation

Estimation of doses to the public are based on the results from an investigation performed in 1987 to 1988. The purpose was to study the dispersion of uranium isotopes (^{234}U, ^{235}U and ^{238}U) to the environment through the smoke stack at the nuclear fuel factory. Meteorological data as well as dry and wet deposition was taken into account when evaluating input data to the plume model code LUCIFER. Results show that doses to the public by inhaled air are well below doses caused by natural background radiation.

3.4. Radiation protection ordinance

SSI regulates the Swedish radiation protection work. Publications from SSI cover both ionising radiation as well as non-ionising radiation. The radiation protection ordinances are to some extent written on the basis of ICRP recommendations.

Table VI describes the ordinances published by SSI and Table VII the internal ambitions at Westinghouse Atom. The latter shows that the ambition by Westinghouse lays 5 mSv below the annual dose limit of 20 mSv recommended according to ICRP 60. Planning doses (see Table VIII) are used when Westinghouse staff travel to other nuclear installations to do service jobs.

TABLE VI. GENERAL RULES ON THE BASIS OF SSI ORDINANCES

Dose criteria	**Dose limit** (mSv)	**Limit of intake** (ALI)	**Comment**
Annual dose limit	50	1	
Cumulated 5-year dose	100	2	On the basis of ICRP 60.
Fertile women	10 / 2 months	0.2	Head of unit shall be informed when pregnancy is stated.
Pregnant women (Foetus dose)	5 / year	0.01 (The mother)	Work is always offered in an area without ionising radiation.

TABLE VII. WESTINGHOUSE ATOM GUIDELINES FOR WORK WITH FUEL MANUFACTURING

Dose criteria	External dose (mSv)	Internal dose (ALI)	Comment ALI = Annual Limit of Intake. (1600 Bq uranium)
Annual dose limit	15	0.3	1 year with 0.6 Bq uranium dust/m^3 air gives 1 ALI corresponding to 30 Bq U-235 lung uptake.
Cumulated 5-year dose	75	1.5	On the basis of ICRP 60.
Fertile women	5 / 2 months	0.1	Head of unit shall be informed when pregnancy is stated.
Pregnant women (Foetus dose)	1 / year	[a]	Work is always offered in an area without ionising radiation.
Indication	> 10 / year > 3 / year	> 0.2 (5 Bq ^{235}U)	Exposure without a cause above this limit must always be followed up.
Measure	> 15 / year	> 0.3 (10 Bq ^{235}U)	Further work with radiation the rest of the year is only permitted in special cases.

[a] not specified

TABLE VIII. SPECIAL GUIDELINES FOR WORK IN NUCLEAR CONSTRUCTION OTHER THAN WESTINGHOUSE ATOM FUEL FACTORY

Dose criteria	**External dose** (mSv)	**Internal dose** (ALI)	**Comment** ALI = Annual Limit of Intake.
Annual dose limit	15	0.3	
Cumulated 5-year dose	75	1.5	On the basis of ICRP 60.
Fertile women	5 / 2 months	0.1	Head of unit shall be informed when pregnancy is stated.
Pregnant women (Foetus dose)	1 / year	-[a]	Work is always offered in an area without ionising radiation.
Indication	> 10 / year > 3 / year	> 0.2	Exposure without a cause above this limit must always be followed up.
Measure	> 15 / year	> 0.3	Further work during the year with uranium or in nuclear construction is only allowed with special permit.
Personal planning dose	< 5 mSv/work	< 0.1	Used for budgets and planning for special works in nuclear construction.
Personal emergency dose	< 3 mSv/day	not applicable	See planning dose

[a] not specified

4. OPERATOR EXPOSURE

The Tables IX and X show the operator exposure during the last nine respectively seven years. As can be seen from Table IX, there is a gap in the collective dose between the years 1997 and 1998. This is due to truncation of the lower doses. Before 1998, we did not consider doses below 0.1 mSv to be reportable to our national dose registry. However, that changed in 1998. In addition to this, we started to use a dosimeter to correct for the background dose. The background was somewhat lower than we expected which lead slight higher doses. These two factors combined account for the drastic increase in collective dose. The Swedish authorities have been informed about this and accepted our explanation

TABLE IX. OPERATOR EXPOSURE DOSES THE LAST NINE YEARS

	Average whole body dose (mSv)	**Collective whole body exposure** (mmanSv)
1992	0.6	59.7
1993	0.6	42.5
1994	0.64	30.7
1995	0.78	62.5
1996	0.80	59.6
1997	0.60	65.4
1998	0.95	140.6
1999	1.28	166.2
2000	1.22	157.0

TABLE X. WHOLE BODY EXPOSURE DOSES TO EMPLOYEES DURING THE LAST SEVEN YEARS

Exposure range	**Number of workers in given range**						
	1994	1995	1996	11997	1998	1999	2000
0.1-0.3	22	38	32	51	57	46	45
0.4-1.0	17	18	29	34	36	31	30
1.1-2.5	8	19	16	20	42	29	29
2.6-5.0	1	5	3	1	13	22	22
>5.0	0	0	0	0	0	2	1

UNITED KINGDOM

1. DESCRIPTION OF THE UK FUEL BUSINESS OPERATIONS

BNFL's Fuel Manufacture and Reactor Services Group: Westinghouse, operates the UK fuel Business manufacturing site at Springfields, near Preston in Lancashire, produces fuel and intermediate fuel products for the nuclear industry in the UK and abroad. Uranium ore concentrates are received on site and are processed to either uranium metal for use in Magnox reactors or to uranium hexafluoride. The latter is sent for enrichment at Capenhurst or abroad. Enriched uranium hexafluoride is also received on site and is converted to oxide fuel or intermediate products for use in Advanced Gas Cooled Reactors or Light Water Reactors. Westinghouse's UK Fuel Business completed commissioning of the Oxide Fuels Complex (OFC) in 1998, which converts enriched uranium hexafluoride to powder, pellets and finished fuel pins under one roof. This is replacing the current facilities on the site which are currently being decommissioned.

1.1 Storage of uranium ore concentrates

A modern purpose-built facility on the Springfields site provides indoor storage conditions for more than 40 000 drums of uranium ore concentrates (UOC), received from all over the world.

1.2 Natural UF_6 conversion

Conversion of UOC to natural UF_6 is a vital step in the production of oxide fuels. A multistage route is used which is designed to produce extremely pure UF_6 from all grades of UOC. The stages comprise UOC purification, conversion to UF_4 and conversion to UF_6.

UF_6 has been manufactured at Springfields for over 40 years and has been supplied for toll enrichment since 1969. Westinghouse UK Fuel Business delivers to commercial enrichment organisations worldwide.

1.3 Conversion of UOC to UF_4

The UOC passes through several chemical processes: dissolution in nitric acid, purification by solvent extraction, concentration and thermal decomposition to produce pure uranium trioxide (UO_3). Conversion of UO_3 to UF_4 then takes place in an efficient rotary kiln plant developed at Springfields in the early 1970s. The two step process involves the reduction of hydrated UO_3 to UO_2 and the hydrofluorination of UO_2 to UF_4. At this point the process route diverges and UF_4 from the rotary kiln is transferred to either the UF_6 conversion plant or the uranium metal production plant, dependant on the type of fuel being manufactured.

1.4 Conversion of UF_4 to natural UF_6

The final stage of the UF_6 process — the conversion from UF_4 to natural UF_6 — is achieved by reacting the UF_4 with elemental fluorine in a fluidised bed reactor. The fluorine is generated on the site using electrolysis cell technology developed at Springfields.

1.5 Magnox fuel production

Westinghouse UK Fuel Business manufactures natural uranium metal fuel clad in Magnox (magnesium alloy) fuel cans for the first generation of UK nuclear power stations, and for the Tokai Mura station in Japan. Fuel element designs for the Magnox stations, although basically similar, are specific to each station. The fuel manufacturing facilities must therefore be readily adaptable and able to manufacture several types simultaneously. More than four million fuel elements with an achieved in-reactor failure rate of less than one in 10 000 have been produced to date.

1.6 Oxide fuel production

Following the enrichment process, the production of ceramic grade UO_2 powder forms an important part of the nuclear fuel cycle. Westinghouse UK Fuel Business has developed the Integrated Dry Route process for converting enriched uranium hexafluoride into UO_2. This is a single kiln process where UF_6 undergoes reduction to UF_4 with hydrogen, followed by hydrolysis of UF_4 with steam. Westinghouse UK Fuel Business experience in the use of the IDR process for UO_2 production spans more than two decades. More than 8 000 tU of IDR powder has been produced at the Springfields site. The single stage kiln process can be used to make a wide variety of fuels including PWR, AGR and BWR. The Oxide Fuels Complex (OFC), houses a total AGR and PWR fuel production capability under one roof.

TABLE I. MANUFACTURING CAPABILITIES OF OFC TOTAL THROUGHPUT

Product	**Capability** (tU/a)
UO_2 Powder	720
AGR finished fuel	290
LWR finished Fuel	300–350

2. REGULATION PROCEDURE IN ENGLAND

2.1 Health, safety and environmental legislation

In the UK operators of nuclear plants must, like their counterparts in other industries, conform to the general health and safety standards in the Health and Safety ant Work Act 1974 (HSW).

The fundamental duty is placed on employers to ensure, so far as reasonably practicable, the health, safety and welfare at work of all their employees. There is also a duty, again, ‘as far asreasonably practicable’ that persons not in their employment are not exposed to risks to their health and safety as a result of activities undertaken. This entails ensuring and demonstrating that risks have been reduced to a level which is ‘as low as reasonably practicable’ (ALARP).

Under the NI Act, apart from certain exemptions, no site may be used for the purpose of installing or operation of a nuclear installation unless a licence has been granted by the Health and Safety Executive (HSE).

Relevant plants (including Springfields) must also comply with the Nuclear Installations Act 1965 (as amended). The NI Act places an absolute liability on the licensee as regards injury to persons or damage to property arising from a nuclear occurrence without proof of fault on the licencee's part.

HM Nuclear Installations Inspectors are appointed under the HSW Act. They administer the NI Act and deal with nuclear and radiological safety issues at licenced nuclear sites. The Nuclear Installations Inspectorate (NII), which are part of the HSE, monitors and regulates the nuclear and radiological safety aspects of an installation by means of its powers under the HSW Act and the licensing procedures. Non nuclear health and safety aspects are primarily the concern of Inspectors from HSE's HM Inspectors of Factories.

Requirements for the protection of the environment and the authorisation of discharges of radioactive waste from nuclear licensed site are regulated by the Environment Agency (EA), in England and Wales, and by the Scottish Environmental Protection Agency in Scotland. Certificates of authorisation issued by these departments impose strict conditions on the discharge and disposal of radioactive waste. They require the operator to carry out detailed environmental monitoring programmes and to use 'Best Practicable Means' (BPM) to further limit the radioactivity of the waste discharged. These authorisations are reviewed on a regular basis.

Discharges to the environment of non-radioactive waste are also subject to regulations under the Environmental Protection Act 1990 (EPA 90). These regulations are enforced by the Environment Agency and cover aerial and liquid discharges and solid waste disposal. The EPA 90 introduced the concept of Integrated Pollution Control (IPC) . IPC is designed to prevent pollution from prescribed processes and substances. Operators are required to select the Best Practicable Environmental Option (BPEO) for plant design and operation and to use the Best Available Techniques Not Entailing Excessive Costs (BATNEEC) to minimise pollution. For discharges made to the aquatic environment, not covered by IPC, the EA issues consents which limit discharges of harmful substances and enable the water quality of the river comply with Environmental Quality Standards. The EPA 90 also introduced new legislation controlling waste collection and disposal of non-radioactive waste on land. It imposes a formal Duty of Care on producers and handlers of waste and also requires that operators of disposal facilities have Waste Management Licences.

3. ENVIRONMENTAL DISCHARGES FROM SPRINGFIELDS

3.1 Radioactive discharges and disposals

Radioactive discharges environmental effect is reported in terms of 'Critical Group Dose'. UK practice is to define the critical group (possibly hypothetical) as a group of people who are representative of those individuals in the population expected to receive the highest dose.

There is no distinct critical group for liquid discharges from the Springfields site. One Houseboat dweller, anglers and wildfowlers receive similar doses. Annual variations in the impact of Springfields' discharges on these groups mean that any one of these groups could be the 'main' critical group in any one particular year. In 1999, houseboat dwellers formed this group with a dose of 0.010 mSv. The dose to the most exposed members of the critical group from aerial discharges is usually below 0.02 mSv. Direct radiation from the site is of very little significance in terms of exposure to the public.

For the last five years the figures for critical group doses received by people due to the operation of the Springfields plant were:

Aerial Discharges for all years 1993–2000 dose to the critical group was <0.02 mSv

Direct Radiation for all years 1993–2000 dose to the critical group was <0.008 mSv

TABLE II. CRITICAL GROUP DOSES

	Critical Group Dose mSv				
	1996	1997	1998	1999	2000
Pathwa					
Boat Dwell in[a]	0.180	0.17	0.11	0.10	0.10
Wildfowling[b]	0.023	0.02	0.023	0.017	0.017
Anglers[c]	0.044	0.016	0.004	0.003	0.003

Notes:
a) approx 90% of this dose is due to discharges form Sellafield
b) approx 50% of this dose is due to discharges from Sellafield
c) approx 30% of this dose is due to discharges from Sellafield

The plants at Springfields involved in the manufacture of oxide fuel contribute to the critical group doses as follows:

Liquid discharges <0.005 mSv — predominately from process residues recovery

TABLE III. AERIAL DISCHARGES:

Year	Critical Group Dose mSv/a	
	Existing plants	OFC
1994	0.0037	<0.000003
1995	0.0030	0.000006
1996	0.0030	<0.000010
1997	0.0030	<0.000010
1998	0.00023	0.000008
1999	0.00009	0.000068
2000	0.00019	0.000016

Springfields discharge radioactive liquid effluent directly into the River Ribble. The critical group for liquid discharges therefore are those people who tend to spend a significant amount of time on intertidal sediments of the river. The houseboat dweller has their boat moored in a muddy creek in the Ribble Estuary receives their dose from direct gamma radiation from radionuclides present in the silt. The majority of this dose comes from ceasium-137 present in historical discharges from Sellafield. The wildfowlers and anglers receive effective dose from beta irradiation of the gonads and skin (about 50%) as well as direct gamma irradiation.

The purification of UOC produces what is known as a raffinate. This raffinate contains daughter products of uranium and thorium present in the UOC. It is these daughter products, together with the parent thorium-232, which are responsible for virtually all of the radioactive

environmental impact of Springfields liquid discharges. The beta activity in the effluent is almost all from the thorium-234 and protoactinium-234m daughters of uranium. The beta activity produced is directly proportional to the amount of uranium processed. The alpha activity is from three main sources; uranium, thoruim-232 and thorium-230.

Aerial discharges from the Springfields site arise from many different sources on site. The main contribution to the critical group dose (via inhalation) are those stacks discharging lung class W material.

3.2 Non Radioactive discharges and disposals

Westinghouse UK Fuel Business Group have monitored non-radioactive discharges for many years. Below are typical monthly discharge figures.

TABLE IV. LIQUID DISCHARGES TO THE RIVER RIBBLE

Trade effluent (Flow: *51 500* m^3) per month				
Substance	Typical monthly discharge (tonnes)	Monthly Limit (tonnes)	Av. discharge cone ($mg.1^{-1}$)	Limit ($mg.1^{-1}$)
Arsenic	0.01	0.2	0.28	3
Cadmium	0.0007	0.007	0.008	0.1
Chromium	0.012	0.15	0.24	2
Copper	0.038	0.4	0.75	5.5
Iron	0.98	5	19.5	80
Lead	0.005	0.2	0.1	2.5
Mercury	0.0002	0.0035	0.005	0.05
Nickel	0.03	0.2	0.5	2.5
Uranium	0.14	0.5	2.83	20
Vanadium	0.02	0.4	0.39	5
Zinc	0.03	0.25	0.65	3.5
Ammonia	2.4	6	100	300
Nitrate	183	385	3 735	N/A
COD	5.7	15	120	
Suspended Solids	14	174.2	979	2 500

Site drainage (Flow: *83 310* m^3) per month				
Substance	Typical monthly discharge (tonnes)	Monthly Limit (tonnes)	Av. discharge cone ($mg.1^{-1}$)	Limit ($mg.1^{-1}$)
Arsenic	0.001	0.005	0.012	0.05
Cadmium	0.000012	0.001	0.00012	0.002
Chromium	0.00022	0.005	0.0025	0.05
Copper	0.00044	0.005	0.006	0.05
Iron	0.123	0.35	1.46	2.5
Lead	0.00028	0.001	0.003	0.02
Mercury	0.00002	0.0005	0.0003	0.005
Nickel	0.00036	0.005	0.04	0.05
Uranium	0.014	0.03	0.18	2
Vanadium	0.00084	0.005	0.01	0.05
Zinc	0.0033	0.02	0.037	0.2

TABLE V. AERIAL DISCHARGES

Discharge Point	Substance	Typical discharge conc. $mg.m^{-3}$)	Limit (mg.rri 3)
Fluorine Production No. 1 Stack	Fluoride	0.27	5
Fluorine Production No. 2 Stack	Fluoride	0.03	5
Line 4 Uranium Hexafluoride Production Plant	Fluoride	0.04	5
New Oxide Fuel Complex Kiln Scrubber	Fluoride	0.05	5
New Oxide Fuel Complex Uranium Hexafluoride Scrubber	Fluoride	0.05	5
Nitric Acid Recove Plant	NO,	250	2300
Enriched Uranium Residues Recovery Plant	NOX	230	750

4. OPERATOR EXPOSURE

TABLE VI. OXIDE FUEL MANUFACTURING EXISTING PLANTS-OCCUPATIONAL DOSE DATA (CLOSED 1999)

Year	Mean Dose mSv	Collective Dose ManSv	Numbers in Dose Range mSv				
			0-2.5	2.5-5	5-10	10-15	>15
1992	5.8	2.8	102	100	226	49	0
1993	3.7	1.9	184	199	132	1	0
1994	3.0	1.3	151	187	75	0	0
1995	2.8	0.94	146	172	17	1	0
1996	2.3	0.80	207	136	14	0	0
1997	2.3	0.69	173	111	14	0	0
1998	1.6	0.35	175	39	1	0	0

TABLE VII. OXIDE FUEL MANUFACTURING OXIDE FUEL COMPLEX-OCCUPATIONAL DOSE DATA

Year	Mean Dose mSv	Collective Dose ManSv	Numbers in Dose Range mSv				
			0-2.5	2.5-5	5-10	10-15	>15
1994	0.6	0.16	255	12	0	0	0
1995	0.3	0.09	296	0	0	0	0
1996	0.34	0.10	294	10	0	0	0
1997	0.78	0.26	320	13	3	0	0
1998	0.90	0.33	345	9	0	0	0
1999	0.89	0.29	323	6	0	0	0
2000	1.09	0.36	321	6	0	0	0

TABLE VIII. OPERATOR EXPOSURE FOR THE WHOLE SPRINGFIELD SITE

YEAR	Average whole body exposure mSv	Collective whole body exposure man-sieverts
1993	1.5	4.3
1994	1.3	3.5
1995	1.1	2.6
1996	1.0	2.2
1997	1.0	2.13
1998	0.85	1.7
1999	0.9	1.5
2000	0.9	1.5

5. COUNTRY PROFILE OF UNITED KINGDOM

Around 25% of the UK's electricity is generated by nuclear power.

A complete fuel cycle is provided by BNFL in the UK for the home market and export.

Mining

No mining of uranium ore takes place in the UK.

Milling

No milling takes place in the UK.

Conversion

Westinghouse operates a conversion facility at its Springfields plant near Preston. Ore is converted to UF_6 for customers. The conversion plant has a hex conversion capacity of 7 050 te as UF_6 per year.

Enrichment

URENCO operates a commercial centrifugal enrichment plant at Capenhurst. This plant has a capacity of 800 t U/y.

Fabrication

Westinghouse Fuel Business Group located at Springfields fabricates a number of different types of fuel. Current production capacities are;

Magnox 1 500 tU/y
AGR 290 tU/y
LWR 330 tU/y

BNFL have sucessfully operated a small scale MOX fuel demonstration facility which has a capacity of 8 tHM/y at Sellafield. BNFL's commercial scale MOX plant is currently undergoing commissioning and will have a capacity of 120 tHM/y.

Spent Fuel Management

BNFL operates a Magnox fuel reprocessing plant at Sellafield with an operational capacity of 1 500tU/y. The Thermal Oxide Reprocessing Plant (Thorp) at Sellafield has an operational capacity of 1 200 tU/y.

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GLOSSARY

activity

the rate of disintegrations per second; 1 disintegration = 1 becquerel.

abdorbed dose

the measurement of radiation absorbed dose (rad) represents the amount of energy deposited per unit mass of absorbing material; 1 rad = 10^{-2} joule/kg = 10^{-2} Gy.

back end (of the nuclear fuel cycle)

refers to operations performed on spent fuel; these include storage in water pools or dry storage facilities, reprocessing and/or disposal, recycling the products of reprocessing (e.g. Pu and U) and waste management.

becquerel (Bq)

becquerel is the measure for radioactivity, i.e. the number of radioactive atoms disintegrating per unit of time.

biosphere

the portion of the Earth's environment normally inhabited by living organisms. It comprises those parts of the atmosphere, the hydrosphere (ocean, seas, inland waters and subsurface waters) and the lithosphere normally related to the human habitat or environment.

boiling water reactor

a light water nuclear reactor in which steam is produced in the reactor and passed directly to the turbogenerator..

CANDU

a heavy water cooled and heavy water moderated nuclear reactor (Canadian deuterium uranium reactor), which uses natural uranium as fuel.

collective dose

the collective dose takes account of the number of people exposed to a source by multiplying the average dose to the exposed group from the source by the number of individuals in the group. If several groups are involved, the total collective quantity is the sum of the collective quantities for each group. The unit of the collective dose is the man sievert (man Sv).

critical group

for a given radiation source and given exposure pathway, a group of members of the public whose exposure is reasonable homogeneous and is typical of individuals receiving the highest dose through the given pathway from the given source.

criticality

the state of a nuclear chain-reacting medium when the chain reaction is just self-sustaining (or critical), i.e. when the reactivity is zero. Often used, slightly more loosely, to refer to states in which the reactivity is greater than zero.

curie (Ci)

former unit of activity, equal to 3.7 × 1010 disintegrations per second; superseded by the becquerel (Bq), which is 1 disintegration per second. Originally, the activity of a gram of radium. Occasionally still referred to as 'gram equivalent radium'.

discharge, routine

a planned and controlled release of radionuclides to the environment. Such releases should meet all restrictions imposed by the appropriate regulatory body.

dose

absorbed dose, organ dose, equivalent dose, effective dose, committed equivalent dose. Or committed effective dose, depending on the context. All these quantities have the dimension of energy divided by mass (joule/kg). The modifying adjectives are often omitted when they are not necessary for the defining the quantity of interest.

effective dose

the effective dose the amount of absorbed radiation per unit mass of matter. The unit of the effective dose is the sievert (Sv).

dose equivalent

the measure of biological effect of radiation requires a unit called a quality factor (QF). The quality factor takes into account the different degrees of biological damage produced by equal doses of different types of radiation. 1 Roentgen equivalent man (rem) is the product of the amount of energy absorbed (rad) times the efficiency of radiation in producing damage (QF), 1 rem = 1 rad·QF. For X rays and gamma radiations and most beta, QF = 1. Alpha radiation has a QF of 20 and for neutrons QF ranges from 2 to 11.

effluent

gaseous or liquid radioactive materials which are discharged into the environment.

fast breeder reactor

a nuclear reactor in which fissions are caused by high energy neutrons and which produces more fissile material than it consumes, (hence breeder). Fissile material is produced both in the reactor's core and through neutron capture in fertile material placed around the core (blanket).

fertile material

consists of nuclides which can be converted to fissile nuclides through neutron capture (e.g. 232Th, 238U, 242Pu).

fissile material

consists of nuclides for which it is possible to set up a nuclear chain reaction using neutrons (e.g. ^{233}U, ^{235}U, ^{239}Pu, ^{241}Pu).

front end (of the nuclear fuel cycle)

operations to provide fuel for use in nuclear reactors, including mining, milling, enrichment and fabrication of fuel.

fuel assembly

a geometrical array of fuel rods, pins, plates, etc. held together by structural components such as end fittings. Also called a fuel bundle, fuel cluster and fuel element.

fuel

see nuclear fuel.

fuel pellet

a cylindrical compact of fuel material, generally oxides, highly compressed and subsequently sintered to a ceramic state.

fuel rod

a basic component of fuel fabricated for service in a reactor, comprising fissile and/or fertile material in a pellet form sealed in a metal tube. Also called fuel pin and fuel subassembly.

gas cooled reactor

a nuclear reactor employing a gas as a coolant (e.g. CO2 or He).

gas graphite reactor

a nuclear reactor employing a gas as a coolant and graphite as a moderator.

Gray (Gy)

Gray is the (physical) unit for measuring the dose received, 1 Gy = 1 J/kg.

heavy metal (HM)

initial thorium, uranium and plutonium content of fuel. Weights of fuel which may be in the form of metal, oxide, carbide and nitrite are stated in terms of initial metal content only.

heavy water reactor

a reactor in which the chain reaction is sustained primarily by fission brought about by thermal neutrons, i.e. neutrons which are in thermal equilibrium with the material in which they are moving. Such reactors use heavy water as a moderator to slow down the neutrons produced in fission to thermal energies.

isotope

nuclides having the same atomic number (i.e. the same number of protons in the nucleus) but different mass (hence a different number of neutrons).

light water reactor

a reactor in which the chain reaction is sustained primarily by fission brought about by thermal neutrons, i.e. neutrons which are in thermal equilibrium with the material in which they are moving. Such reactors use light water as a moderator to slow down the neutrons produced in fission to thermal energies.

mixed oxide fuel (MOX)

fuel manufactured from oxides of both uranium and plutonium.

moderator

a component (e.g. water, graphite) that slows high energy neutrons.

nuclear facility

a facility and its associated land, buildings and equipment in which radioactive materials are produced, processed used, handled stored or disposed of on such a scale that consideration of safety is required.

nuclear fuel

fissionable and fertile material used in a nuclear reactor for the purpose of generating energy. Usually, the material is in the form of pellets, which are stacked and sealed into a metal rod. The rods are grouped together in a fuel assembly.

nuclear fuel cycle

all operations associated with the production of nuclear energy, including sequentially mining, milling, and enrichment of uranium; fabrication of fuel; operation of nuclear reactors; optional reprocessing of spent fuel; recycling the products of reprocessing (e.g. Pu and U), decommissioning; any activity for

radioactive waste management and any research or development activity related to any of the foregoing.

nuclear fuel cycle, closed

the closed fuel cycle concept involves the recycling of fissile and fertile material recovered from the reprocessing of spent fuel. Interim storage may be required.

nuclear fuel cycle, once-through

the once-through fuel cycle concept involves the disposal of the spent fuel following its use in the reactor. Interim storage is likely to be required.

nuclear fuel fabrication facility

a (nuclear) facility where the nuclear material is fabricated into fuel elements for use in a nuclear reactor.

nuclear reactor

a heat engine configured to sustain a controlled nuclear chain reaction when fuelled with fissionable materials. Fissions may be caused by relatively low energy (thermal) neutrons or by high energy (fast) neutrons, hence classified as thermal or fast reactors.

nuclear power reactor

a nuclear reactor designed to produce electricity.

nuclear safety

the conditions established by the systematic analysis and reduction of risks associated with the operation of a nuclear fuel cycle facility and its services..

operation

all activities performed to achieve the purpose for which the nuclear facility was constructed, including maintenance, refuelling, in-service inspection and other associated activities.

rad

rad is former unit for radiation absorbed dose, i.e. the amount of energy deposited per unit mass of absorbing material, 1 rad = 10-2 Gy.

radiation exposure

the measurement of radiation exposure in air as ionizations per unit mass of air due to X ray or gamma radiation. 1 Roentgen (R) = 2.58·10-4 Coulomb/kg air.

radioactive isotopes (or radionuclide)

unstable isotopes which undergo spontaneous change, i.e. radioactive disintegration or radioactive decay, at definite rates. The process is accompanied by the emission of one or more types of radiation, such as alpha particles, beta particles and gamma rays and results in the formation of new nuclides. Isotopes (or nuclei) are distinguished by their mass and atomic number.

radioactive effluent

see effluent.

radioactivity

property of certain nuclide to undergo spontaneous disintegration in which energy is liberated, generally resulting in the formation of new nuclides. The process is accompanied by the emission of one or more types of radiation, such as alpha particles, beta particles and gamma rays. Becquerel is the measure for radioactivity, i.e. the number of radioactive atoms disintegrating per unit of time.

rem

rem is the former unit of dose equivalent, 1 rem = 10^{-2} Sv.

Sievert (Sv)

Sievert is the (biological) unit for dose equivalent, i.e. measuring the dose absorbed by tissues, taking into account the radiation type, 1 Sv = 1 J/kg.

thermal reactor

a nuclear reactor in which high energy (fast) neutrons produced by the fission process are slowed down to a low energy (thermal) through a moderator. The thermal neutrons in turn cause fissions in the fissile material.

Source: [29–32]

ABBREVIATIONS

ABB	ASEA Brown Bovery
ADU	ammonium diuranate
AGR	advanced gas cooled reactor
ALARA	as low as reasonably achievable
ALARP	as low as reasonably pPractical
ALI	annual limit of intake
ATR	advanced thermal reactor
AUC	ammonium uranyl carbonate
BNI	nuclear material handling facility (French abbreviation)
BPM	best practical means
BWR	boiling water reactor
CANDU	Canadia deuterium-uranium reactor
COD	chemical oxygen demand
FBR	fast breeder reactor
GCR	gas cooled, graphite-moderated reactor
HM	heavy metal
HWR	heavy water reactor
IDR	integrated dry route
IPC	integrated pollution control
LMFBR	liquid metal cooled fast breeder reactor
LWGR	light water cooled, graphite moderated reactor
LWR	light water reactor
Magnox	gas cooled reactor mostly used in the UK with magnesium oxide fuel
MOX	mixed oxide
NI	nuclear installations
PHWR	pressurized heavy water reactor
PWR	pressurized water reactor
RBMK	light water cooled reactor of Russian type
SFS	Swedish acts, related to nuclear activities
SS	suspended solids
UOC	uranium ore concentrate
WWER	pressurized water reactor of Russian type

ORGANIZATIONS

BMU	German Federal Ministry for the Environment, Nature Conservation and Nuclear Safety
CSN	Consejon de Seguridad Nuclear, Spanish Regulation Authority
DSIN	French Safety Authority
EA	Environmental Agency
HSE	United Kingdom Health and Safety Executive
HSW	United Kingdom Health and Safety Work Act
IAEA	International Atomic Energy Agency
ICRP	International Commission on Radiological Protection
JNC	Japan Nuclear Cycle Development Institute
NII	United Kingdom Nuclear Installations Inspectorate
OFC	United Kingdom Oxide Fuels Complex
SKI	Swedish Nuclear Power Inspectorate
SSI	Swedish Radiation Protection Institute
TÜV	German Technical Inspection Agencies

UNITS/CONVERSIONS

1 curie (Ci)	=	$3.7 \cdot 10^{10}$ becquerel (Bq)
1 becquerel (Bq)	=	1 disintegration/s
1 Gray (Gy)	=	1 joule/kg (J/kg)
1 rad	=	$1 \cdot 10^{-2}$ joule/kg or 10^{-2} Gy
1 rem	=	1 rad·QF, (QF = quality factor) or 10^{-2} Sv
1 Roentgen	=	$2.58 \cdot 10^{-2}$ Coulomb/kg air
1 sievert (Sv)	=	1 joule/kg
1 t HM	=	1 metric tonne of heavy metal
1 t U	=	1 metric tonne of uranium

Prefixes

Symbol	Name	Factor
E	exa	10^{18}
P	peta	10^{15}
T	tera	10^{12}
G	giga	10^{9}
M	mega	10^{6}
k	kilo	10^{3}
h	hecto	10^{2}
da	deca	10^{1}
d	deci	10^{-1}
c	centi	10^{-2}
m	milli	10^{-3}
μ	micro	10^{-6}
η	nano	10^{-9}
p	pico	10^{-12}
f	femto	10^{-15}
a	atto	10^{-18}

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