



# ***Status and trends in spent fuel reprocessing***

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## FOREWORD

Spent fuel management has always been an important part of the nuclear fuel cycle and is still one of the most important activities in all countries exploiting the peaceful use of nuclear energy. Continuous attention is being given by the IAEA to the collection, analysis and exchange of information on spent fuel management. Its role in this area is to provide a forum for exchanging information and to co-ordinate and encourage closer co-operation among Member States in certain research and developing activities that are of common interest.

As part of spent fuel management, reprocessing activities have been reviewed from time to time on a low profile level under the terminology 'spent fuel treatment'. However, spent fuel treatment covers, in broad terms, spent fuel storage (short, interim and long term), fuel rod consolidation, reprocessing and, in case the once-through cycle is selected, conditioning of the spent fuel for disposal. Hence the reprocessing activities under the heading 'spent fuel treatment' were somewhat misleading.

Several meetings on spent fuel treatment have been organized during the last decade: an Advisory Group meeting (AGM) in 1992, a Technical Committee meeting in 1995 and recently an Advisory Group meeting from 7 to 10 September 1998. The objectives of the meetings were to review the status and trends of spent fuel reprocessing, to discuss the environmental impact and safety aspects of reprocessing facilities and to define the most important issues in this field. Notwithstanding the fact that the Summary of the report does not include aspects of military reprocessing, some of the national presentations do refer to some relevant aspects (e.g. experience, fissile stockpiles).

During the last Advisory Group meeting, the participants reviewed and updated an unpublished draft report on the state of the art on reactor fuel reprocessing, prepared by the Technical Committee in 1995. The AGM recommended that the updated report together with the national presentations made to the 1998 AGM should be published, with appropriate acknowledgement of the valuable work of the 1995 Technical Committee.

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## SUMMARY

### 1. INTRODUCTION

Three options for spent fuel management exist at present: a once-through cycle requiring the direct disposal of spent fuel; a closed cycle with reprocessing of spent fuel, recycling of recovered plutonium and uranium and the disposal of associated waste; and deferral of the decision which of the previous options to choose. Recycling includes the fabrication of MOX (a mixture of the oxides of uranium and plutonium) and reprocessed uranium fuel, with subsequent irradiation in reactors.

In order to allow deferral of the decision, the spent fuel has to be kept in an interim storage facility, where it can be monitored continuously until the decision is taken while actively evaluating the strategies. The selection of spent fuel strategy is a complex decision with many factors to be taken into account, covering politics, economics, resource conservation and environmental protection.

According to IAEA data, at present there are more than twenty five countries with well developed nuclear power generation programmes [1]. Of these about seventeen are directly or indirectly involved in civil reprocessing. The countries with available civil reprocessing facilities are France, India, Japan, Russia and the United Kingdom, with China currently building a facility. An overview of the history of reprocessing activities since the development of nuclear energy can be found in [2].

Originally, the objective of the closed fuel cycle concept was to recycle the separated plutonium and uranium in fast breeder reactors (FBRs). The penetration of fast breeders however, has not been as rapid as envisaged a few decades ago. At present, although most of the existing nuclear reactors in the world are thermal, such reactors themselves have a demonstrated potential for recycling uranium and plutonium in substantial quantities. Currently, thermal recycling of plutonium as MOX fuel is being carried out in pressurized water reactors (PWRs) and boiling water reactors (BWRs) in Belgium, France, Germany, Japan and Switzerland. Thermal recycling of uranium is being carried out in PWRs in Belgium and France, in RBMKs (Reaktor Bolshoy Moschnosti Kipyaschiy, Russian type of water cooled, graphite moderated reactors) in the Russian Federation and has been carried out in AGRs (advanced gas cooled reactors) in the UK.

### 2. REPROCESSING AND RECYCLING

Reprocessing as currently conducted involves shearing, dissolution and uranium and plutonium recovery by liquid-liquid extraction using the PUREX process. Spent fuel reprocessing meets today's requirement of natural resource conservation and reduction of waste toxicity, and it is planned that reprocessing will soon be in balance with MOX fabrication and recycle capacity. MOX fuel is being used on a commercial basis in thermal reactors by many utilities, although it is acknowledged that FBRs would be more efficient in generating electricity from recycling the recovered plutonium and uranium.

The number of thermal reactors licensed to load MOX fuel is increasing and it is expected that by the beginning of the next century, sufficient recycling capability should be available such that the separated civil plutonium inventory could start to be gradually reduced (however, some institutional co-operation would be required). Within several years, the inventory could be reduced to a level determined only by the working stocks of the recycle facilities. It should be mentioned, that this policy will have the additional benefit of avoiding the in-growth of americium in plutonium and the costs of its removal through a purification process. New light water reactors (LWRs), designed to use 100% MOX cores, and new FBRs, specifically designed to breed or consume plutonium, should make it possible in the future to control the levels of production and separation of plutonium to match world demand for electricity.

The recycling of reprocessed uranium (RepU) does not face the same time-expiration issue experienced with plutonium when used in thermal reactors. The utilities, which own RepU, can choose when it is desirable to recycle it in their reactors. It has been demonstrated that in-core behaviour of RepU fuel is very similar to that of natural  $\text{UO}_2$  fuel and such recycle is already being conducted on a

commercial scale. Dedicated fuel cycle facilities to handle RepU are operational or under construction to meet utility needs.

Besides recycling, the conditioning of the waste arising from reprocessing is another important issue. Conditioning of waste is the solidification of liquid waste and the treatment of all solid waste. Reprocessing results in the recovery of more than 99.8% of the uranium and plutonium and consequently reduces the residual toxicity of the waste. Efforts in minimizing waste volumes and their toxicity have consequences for the scale and cost of transportation, interim storage facilities and the repositories required for final disposal.

Conditioning has the aim of providing adequate protection to man and the environment against the radiological hazard associated with waste. The vitrification of the high level waste (HLW) provides a very efficient confinement since the radionuclides are part of the glass matrix. Other methods, such as the confinement of these nuclides in mineralized matrices, are being studied, for instance in Russia, with the idea of selecting a matrix for each nuclide or group of nuclides. Benefits to interim storage facilities as well as to final repositories should result from efforts at the reprocessing facilities to standardize as far as possible the forms of the waste containers themselves.

### 3. STATUS AND TRENDS IN REPROCESSING

#### 3.1. Current status

In 1998, the annual spent fuel arisings from all types of power reactors (i.e. of 349 GWe) world-wide amounted to about 10 500 tonnes of heavy metal (t HM). The total amount of spent fuel accumulated world-wide at the end of 1997 was over 200 000 t HM and projections indicate that the cumulative amount generated in the world by the year 2010 may surpass 340 000 t HM and by the year 2015 395 000 t HM.

There is considerable experience in the civil reprocessing of irradiated fuel on a industrial scale in several countries. The civil reprocessing capacities for irradiated fuel is currently about 5000 t HM/a, see Table I. France is successfully operating reprocessing plants for oxide fuel. It has already reprocessed 1670 t spent fuel in 1997 and more than 12 000 t HM from the start of the La Hague plants, while the United Kingdom's THORP plant, now fully operational, has a capacity of up to 1200 t HM/a and has reprocessed about 1500 t HM of AGR and LWR fuel. The Russian RT-1 plant has a capacity of 400 t HM/a and to date some 4000 t of WWER fuel has been reprocessed. Reprocessing experience in India and Japan is equally relevant although installed plant capacities are not so large. The Tokai reprocessing plant has reprocessed about 1000 t oxide fuel. On a world scale, there is experience in the reprocessing of a total of about 77 000 t HM of irradiated fuel over the last 40 years (see Table II).

As a result of international co-operation and agreement, the French UP3 and the UK's THORP plants undertake reprocessing under contract for other countries, including Japan. Similar arrangements exist between Russia and some Central European countries, although the quantities of fuel involved are not so large. France and the UK are also offering a guaranteed MOX reprocessing service. MOX fuel can be reprocessed in the same plants as for LWR UO<sub>2</sub> fuel and it has been demonstrated in France and Japan. No major problems were encountered in the reprocessing of 20 t of such fuel.

International reprocessing contracts have required the satisfactory resolution of issues such as the transport of fuel and the return of products and waste to the countries of origin. These contracts have been subject to the full rigours of international safeguards. The successful return of the waste (and products) arising from such reprocessing has been demonstrated, with the waste forms and packaging conforming to specifications agreed with the safety regulators in the countries where the fuel originated.

As an example of the reprocessors efforts to improve the international market for reprocessing by helping customers respond to domestic waste management pressures, the UK is now offering overseas customers the option to substitute a small quantity of high level waste that is radiologically equivalent to

the larger volume of intermediate level waste that would otherwise be returned to them under contractual agreements. Such substitution would result in large resource savings and reduced waste transport. With similar objectives, France is planning to return waste to customers in single form containers that can accept both high and intermediate level waste.

TABLE I. REPROCESSING CAPACITIES IN 1998

Country	Site	Plant	Fuel type				(t HM/a)
			MAGNOX	LWR	FBR	Other	Total
France	La Hague	UP2		800			800
	La Hague	UP3		800			800
India	Trombay	PP				60 <sup>a</sup>	60
	Tarapur	Prefre-1				60 <sup>b</sup>	60
Japan	Tokai-mura	TRP		100			100
Russian Fed.	Chelyabinsk	RT-1		400			400
UK	Sellafield	B205	1 500				1 500
	Sellafield	THORP		1 200 <sup>c</sup>			1 200
	Dounreay	UKAEA RP			10		10
Total			1 500	3 300	10	120	4 930

<sup>a</sup> Research reactor

<sup>c</sup> LWR/AGR

Status yearend 1998

<sup>b</sup> Derated

Note: In India, there is a 100 t HM/a plant under commissioning at Kalpakkam (Prefre-2), see Table III.

TABLE II. CUMULATIVE AMOUNT OF CIVIL REPROCESSED SPENT FUEL

Country	Site	Plant	Fuel type				(t HM)
			GCR	LWR	FBR	MOX	Total
Belgium	Mol	Eurochemic <sup>a</sup>	19 <sup>b</sup>	86			105
France	Marcoule	UP1 <sup>a</sup>	18 000 <sup>c</sup>				18 000
	La Hague	UP2/UP3		12 452	10	9.4	12 471
Germany	Karlsruhe	WAK <sup>a</sup>		180			180
India	Trombay	PP					
	Tarapur	Prefre-1					
Japan	Tokai-mura	TRP		936		10 <sup>d</sup>	946
Russian Fed.	Chelyabinsk	RT-1		4 000			4 000
UK	Sellafield	B205	40 000 <sup>e</sup>				40 000
	Sellafield	THORP		1 500 <sup>f</sup>			1 500
	Dounreay	UKAEA RP			14		14
USA	West Valley	NFS <sup>a</sup>		194			194
		Total	58 019	19 348	24	19	77 410

<sup>a</sup> Closed facility

<sup>d</sup> spent fuel from Fugen

Status yearend 1998

<sup>b</sup> CANDU, GCR and other

<sup>e</sup> Magnox

<sup>c</sup> UNGG

<sup>f</sup> LWR/AGR

### 3.2. Future trends

New reprocessing plants under construction and/or at the planning stage are as follows: China has a pilot reprocessing plant under construction and plans to commission it by the year 2001 and there are further plans for a 800 t HM/a facility by about 2020. Japan's Rokkasho Mura plant for commercial reprocessing at 800 t HM/a is planned for commissioning around 2003 (see Table III).



TABLE III. PAST, CURRENT AND PROJECTED REPROCESSING CAPACITIES

												t HM/a
Country	Site	Plant	Fuel Type	Start-up Year	Year Closed	1998	1999	2000	2005	2010	2015	2020
Belgium	MOL	Eurochemic	LWR	1966	1975							
China	Jiuquan	RPP	LWR	2001					25	25	25	25
	Planned			2020								800
France	Marcoule	APM	FBR	1988	1996							
	Marcoule	UP1	GCR	1958	1997							
	La Hague	UP2	LWR	1976/1994		800	800	800	800	800	800	800
	La Hague	UP3	LWR	1989		800	800	800	800	800	800	800
Germany	Karlsruhe	WAK	LWR	1971	1990							
India	Trombay	PP	Res. Reactor	1964		60	60	60	60	60	60	60
	Tarapur	PREFRE-1	PHWR	1979		60	60	60	60	60	60	60
	Kalpakkam	PREFRE-2	PHWR	1999			100	100	100	100	100	100
	Kalpakkam	PREFRE-3A	PHWR	2003					150	150	150	150
	Tarapur	PREFRE-3B	PHWR	2004					150	150	150	150
Japan	Tokai-mura	PNC TRP	LWR	1977		100	100	100	100	100	100	100
	Rokkasho-mura	RRP	LWR	2003					520	800	800	800
Russia	Chelyabinsk	RT1	WWER-440	1971		400	400	400	400	400	400	400
	Krasnoyarsk	RT2	WWER-1000	2015							1 500	1 500
UK	Dounreay	UKAEA RP	FBR	1980		10	10	10	10	0	0	0
	Sellafield	B205	Magnox	1964		1 500	1 500	1 500	1 500	1 500	0	0
	Sellafield	THORP	LWR/AGR	1994		1 200	1 200	1 200	1 200	1 200	1 200	1 200
USA	West Valley	NFS	LWR	1966	1972							
	Hanford	Rockwell	U metal	1956	1989							
	Savannah River	SR	U metal	1954	1989							
	Idaho Falls	R	U-Al alloy	1959	1992							
Total						4 930	5 030	5 030	5 875	6 145	6 145	6 945

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India has a 100 t HM/a facility under hot commissioning trials at present near Chennai (Madras) and work has started on two plants with a combined capacity of about 300 t HM/a. The RT-2 facility at Krasnoyarsk in Russia is likely to be completed after the year 2015 and will have a capacity of up to 1 500 t HM/a when fully operational. At present, no additional reprocessing facilities are being planned in the UK or in France. The total reprocessing capacity for all fuel types could be up to about 6 000 and 7 000 t HM/a by 2010 and 2020, respectively.

It is unlikely, that in the next 20 years the well known PUREX reprocessing technology will be replaced or fundamentally changed. However, evolutionary steps to achieve process optimisation are being taken, including consideration of continuous rather than batch dissolution, improvements in feed clarification and separation technology, and waste minimisation. Alternative reprocessing technologies are also being examined with a view to reducing costs and environmental impact. Examples of possible such technologies are, molten salt, fluoride volatility and, for application in the DUPIC concept (in which spent PWR fuel undergoes a further irradiation cycle in a PHWR), the aerox process.

The overall driver for these developments is the need to adapt to changing utility management, regulatory and public acceptability requirements, whilst maintaining sustainability in terms of resource, non-proliferation and environmental safety. However, within the overall fuel cycle there are complex interactions between the various stages and their associated costs and rigorous evaluation is needed to ensure that the benefits yielded in one area, such as reprocessing, do not have negative impacts in other areas.

#### 4. PLUTONIUM AND URANIUM RECYCLING

Plutonium has been successfully recycled in the form of MOX fuel in thermal reactors for more than thirty years. In most reactors, this has been done on an experimental basis and the performance of the MOX fuel has been similar to that of uranium fuel. Currently, the use of MOX fuel has been established on an industrial scale in a few countries. In Belgium, France, Germany, Japan and Switzerland a considerable number of the thermal power reactors (PWRs and BWRs) are either licensed (about 40) or have applied for a license (about 13) to use MOX fuel at levels of up to 30% of the reactor core (see Tables IV and V) and 33 reactors have actually loaded MOX fuel in their reactor core. MOX fuel already achieved good irradiation performance. For instance, in France the average burnup of spent MOX fuel is 36 GW·d/t HM, while irradiation up to 44 GW·d/t HM will shortly be reached. In Belgium and Switzerland, the batch average burnup of MOX fuel is currently between 43–46 and 40–44 GW·d/t HM, respectively, which implies that in these countries MOX fuel has the same discharge burnup as U fuel. Because the ratio of fissile plutonium will decrease in high burnup UO<sub>2</sub> spent fuel, fresh MOX fuel assemblies will require higher plutonium content to deliver the same quantity of energy. This will increase the plutonium consumption of the MOX fuelled reactors.

TABLE IV. STATUS OF LARGE SCALE MOX FUEL UTILIZATION IN THERMAL REACTORS  
Status yearend 1998

	Number of Thermal Reactors			
	Operating [1]	Licensed to use MOX FAs <sup>a</sup>	Loaded with MOX FAs <sup>a</sup>	Applied for MOX licence <sup>b</sup>
Belgium	7	2	2	
France	58	20	17	8
Germany	20	12	10	4
Japan	53	3	1	1
Switzerland	5	3	3	
Total	133	40	33	13

<sup>a</sup> There are a number of reactors, notably in Europe and India, not included in this Table, which are licensed to use MOX fuel and have MOX fuel loaded on an experimental basis;

<sup>b</sup> Technically capable reactors planned to be licensed.

TABLE V. THERMAL REACTORS UTILIZATING MOX FUEL ON A LARGE SCALE

Status yearend 1998

	Licensed <sup>a</sup>	Loaded <sup>a</sup>	Applied for licence <sup>b</sup>
Belgium	Doel 3 Tihange 2	Doel 3 Tihange 2	
France	Blayais 1 Blayais 2 Dampierre 1 Dampierre 2 Dampierre 3 Dampierre 4 Gravelines 1 Gravelines 2 Gravelines 3 Gravelines 4 Tricastin 1 Tricastin 2 Tricastin 3 Tricastin 4 Saint-Laurent B1 Saint-Laurent B2 Chinon B1 Chinon B2 Chinon B3 Chinon B4	Blayais 1 Blayais 2 Dampierre 1 Dampierre 2 Dampierre 3 Dampierre 4 Gravelines 1 Gravelines 2 Gravelines 3 Gravelines 4 Tricastin 1 Tricastin 2 Tricastin 3 Tricastin 4 Saint-Laurent B1 Saint-Laurent B2 Chinon B4	Blayais 3 Blayais 4 Cruas 1 Cruas 2 Cruas 3 Cruas 4 Gravelines C5 Gravelines C6
Germany	Brokdorf Grafenrheinfeld Grohnde Gundremmingen B Gundremmingen C Isar 2 Obrigheim Philippsburg 2 Unterweser Neckarwestheim 2 Emsland Neckarwestheim 1	Brokdorf Grafenrheinfeld Grohnde Gundremmingen B Gundremmingen C Isar 2 Obrigheim Philippsburg 2 Unterweser Neckarwestheim 2	Biblis A Biblis B Brunsbüttel Krümmel
Japan	Fugen Takahama 3 Takahama 4	Fugen	Fukushima-Daiichi-3
Switzerland	Beznau 1 Beznau 2 Gösgen-Däniken	Beznau 1 Beznau 2 Gösgen-Däniken	

<sup>a</sup> There are a number of reactors, notably in Europe and India, not included in this Table, which are licensed to use MOX fuel and have loaded MOX fuel on an experimental basis;

<sup>b</sup> Technically capable reactors planned to be licensed.

Those countries with an existing uranium recycling strategy, generally have the necessary fuel cycle facilities available (e.g. conversion and/or enrichment plants) and large scale facilities are either available or under construction to offer uranium recycle services to the international market. Some countries regard reprocessed uranium as a strategic reserve and intend to store it for possible use in the future when availability and /or price of new uranium could be less attractive than at present.

Plants for MOX fuel fabrication are in operation in Belgium, France, Japan, U.K. and India (see Table VI). In the U.K. a large scale MOX fabrication plant has been constructed and is awaiting consent to start operation. In Russia, the first pilot plant (with a capacity of 1 t HM/a) for fabricating MOX fuel is under construction inside the RT-1 plant. A new MOX plant (Complex 300) is planned to commence operation in 2010 [3]. There are plans for the construction of a new MOX plant in Japan and of a demonstration facility in China.

More efficient use of Pu and RepU is made in FBRs, where multiple recycling is possible, and has already been demonstrated. In Russia, it is intended to recycle plutonium in commercial FBRs and there are plans to construct three such reactors.

Experience of the recycling of reprocessed uranium on an industrial scale has been gained in the UK, Russia and France. Several thousands of tons resulting from the reprocessing of spent Magnox fuel at Sellafield have been processed in the UK to produce fresh AGR fuel. In Russia, the uranium coming out of the RT-1 plant at Chelyabinsk has been reused over many years for the fabrication of RBMK fuel. Belgium is recycling all of its RepU in one dedicated PWR and France is using RepU fuel in PWRs on a industrial basis. Other countries, including Japan and China, are recycling uranium on a smaller scale.

TABLE VI. CURRENT AND PLANNED MOX FUEL FABRICATION CAPACITY  
(t HM/a)

Country	Site	Plant	1998	2000	2005	2010
Belgium	Dessel	P0	35	40	40	40
France	Cadarache	CFCa	35	40	40	40
	Marcoule	MELOX	120	200 <sup>a</sup>	200 <sup>a</sup>	250 <sup>a</sup>
India	Tarapur	AFFF	5	10	10	10
Japan	Tokai	PFPF	15 <sup>b</sup>	15 <sup>b</sup>	5 <sup>c</sup>	5 <sup>c</sup>
	Rokkasho-ra	MOX FFF			100	100
Russian Fed.	Chelyabinsk	inside RT-1			1 <sup>d</sup>	1 <sup>d</sup>
	Chelyabinsk	Mayak, Complex 300				10
UK	Sellafield	MDF	8	8	8	8
	Sellafield	SMP		120	120	120
Total			218	433	524	584

<sup>a</sup> date not fixed

<sup>b</sup> for ATR Fugen and FBR Monju

<sup>c</sup> for FBR Monju

<sup>d</sup> for FBR and WWER-1000

Status yearend 1998

## 5. WASTE FROM REPROCESSING

### 5.1. Current status

At present, reprocessing produces two types of waste:

- Waste from the process itself in the form of a liquid solution of fission products and actinides; and
- Waste comprising hulls and end fittings from the structure of the fuel, waste coming from the treatment of process effluents and technological waste from maintenance operations.

The reference strategy for the management of the first type, classified as high level waste (HLW), is vitrification followed by above-ground interim storage and eventual deep geological disposal. Several international studies have confirmed that such a strategy can be implemented with a variety of geological options and with very low environmental impact. There is no immediate need for an alternative approach and future developments are likely to focus on site-specific and waste-form-specific issues relating to final disposal.

There are different approaches being used for the second type, classified as intermediate level waste (ILW). The UK, encapsulates intermediate level waste in cement, whereas France is now storing hulls and end fittings for compaction from 2000 onwards and return of waste to customers in steel containers. In Russia and Japan, such ILW is stored in canisters, uncompacted, prior to possible conditioning in the future to facilitate disposal.

## **5.2. Future trends**

### *5.2.1. Waste minimization*

Waste minimization programmes are currently being actively pursued. The purpose is to reduce volumes and radiotoxicity to facilitate interim storage and final disposal leading to reduced overall costs. Research is also being undertaken to segregate specific nuclides for transmutation with a view to reduce their long-term environmental impact. There are a number of long-term projects in progress on the partitioning and transmutation (P&T) of long lived nuclides (especially actinides) from high level waste.

Prominent among these are the OMEGA project in Japan, the Russian programme, the SPIN project in France, and work in the USA and China. The OECD/NEA has an information exchange programme on P&T, the CEC has funded several projects in the field of new fuel cycle concepts (i.e. strategic studies, partitioning and transmutation techniques) and the IAEA has a few small co-ordinated research projects. In the Russian programme, some nuclides have already been separated successfully at reprocessing plants. For example, the separation of neptunium has been achieved and a facility for the recovery of caesium and strontium has been operational since 1995. An advanced flowsheet for the partitioning of HLW is being planned for implementation in the proposed new reprocessing plant at Krasnoyarsk. However, before such technology could be implemented on a large scale, a number of uncertainties need to be resolved. It would add to the costs of reprocessing and the fate of separated products, that cannot be utilized or subjected to transmutation, needs to be defined.

Research and development work is currently underway to reduce the gaseous discharges to the environment of, for example, Kr-85, I-129, C-14 and tritium.

### *5.2.2. Waste volume reduction*

Ten years of operation of the La Hague plants have demonstrated that it is possible to reduce the amount of waste from 3 to less than 0.5 m<sup>3</sup>/t HM. This reduction has resulted from the discontinuation of bitumen encapsulation and the compaction of hulls and end-fittings. Even without possible further reductions, the volume of waste arising from reprocessing requiring geological disposal is already four times less than the corresponding volume of the spent fuel itself.

## **6. ENVIRONMENT AND SAFETY**

As mentioned above, reprocessing and recycling conserve natural resources and provide environmental benefits by reducing, as compared to the once-through cycle, the volumes of waste requiring treatment and disposal. By separating uranium and plutonium, the closed cycle reduces the amount of radioactive nuclides in the waste to be disposed of and, on time scales relevant to the potential effect on the biosphere, decreases the radiotoxicity of this waste by about an order of magnitude.

Existing reprocessing facilities generate aerial and liquid discharges. All such discharges and plant operations conform to the requirements of the regulatory authorities with appropriate regard to the ALARA principle and the impact from these discharges on the environment is negligible. The operators of reprocessing facilities are required to provide reports of environmental discharges on a routine basis, e.g. see [4].

Experience has shown that the authorized limits on discharges have steadily been reduced. As an example of pressures to reduce discharge levels further, the OSPAR (Oslo-Paris) Convention has

specified reductions in discharges to the marine environment that imply a near zero level by 2020 taking into account technical and radiological impact.

A similar consideration applies for occupational exposure. Limits are set by the regulatory authorities, to whom regular reports are made, and experience shows that the imposed limits have been reduced over the years. Although actual occupational exposure levels in plants are already low compared with natural background radiation, design of new facilities and operating practices anticipate possible further reductions in existing authorized limits.

Other waste arisings from reprocessing plants are stored, treated and packaged as appropriate in anticipation of ultimate disposal. This waste and associated operations, including interim storage, are subject to regulatory controls and the repositories for their ultimate disposal will conform to standards which will ensure that the environmental impact is negligible.

Compliance with non-proliferation requirements is also achieved. For example, commercial reprocessing (UK and France) and MOX fuel fabrication (Belgium and France) facilities are operated under the control of the IAEA and EURATOM. They have demonstrated the security and safeguardability of operations that have involved the handling, storage, transport and recycling of plutonium, conducted variously over periods of up to forty years.

Transport of spent fuel, uranium, plutonium and waste has been carried out safely for over 30 years under international guidelines and involving transport variously by rail, road, sea and air.

Conventional safety in reprocessing plants cannot be overlooked and there is an evolution of practices to conform with world class standards in conventional plants. In a number of countries, reprocessing plants already operate to or are planning to achieve relevant ISO standards, e.g. the 9000 and 14 000 series.

## 7. CONCLUSIONS

The reprocessing of irradiated fuels from power reactors is a mature, commercially and industrially proven technology. According to what has been recorded in 1998, nearly 5000 t HM/a capacity is already available and this will increase up to nearly 6000 and 7000 t HM/a by the year 2010 and 2020, respectively. Transport of fuel and waste is performed regularly with an exemplary safety record over some 30 years of operation. Recovered fissile material and conditioned waste are meeting the specifications of the appropriate regulatory authorities.

Reprocessing plants are performing well and the recycle capacity is increasing with the expansion of MOX fabrication and MOX loading capability. A balance between recycling and reprocessing activities will allow inventories of separated plutonium to be controlled and minimised to levels consistent with process requirements.

Industrial experience over some forty years has demonstrated that commercial reprocessing and recycling, managed under international safeguards, need not be viewed as a proliferation threat.

There has been a continuing reduction in the volume of waste arisings from reprocessing. This trend will continue with the implementation of improved technology and operating practices. R&D programmes to study the partitioning and transmutation of environmentally-significant radionuclides are being pursued to further enhance the effectiveness of waste-minimisation programmes. There are continuing opportunities to optimize the treatment and disposal of radioactive waste.

In summary, the closed fuel cycle is a proven and mature industrial technology which provides the means for effective utilization of nuclear fuel whilst meeting strict environmental standards.

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# CHINA'S SPENT FUEL TREATMENT: THE PRESENT STATUS AND PROSPECTS



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## Abstract

In the mid 1980s, China launched the development of nuclear power dominated by PWRs and opted for the closed fuel cycle strategy. On the basis of irradiated fuel reprocessing for defence purpose, an R&D programme for civil reprocessing has been implemented. Currently, China's spent fuel arising is limited but its amount will sharply increase with nuclear power expansion early next century. Spent fuel stored at reactor site for at least 5 years will be transported either by a combination of sea and rail or by road directly to the Lanzhou Nuclear Fuel Complex. A wet centralized storage facility with a 550 tHM capacity has been built for interim storage of spent fuel. Also, a multi-purpose reprocessing pilot plant with a maximum throughput of 400 kg HM/d is now under construction and will be put into commissioning by the turn of the century. A large-scale commercial reprocessing plant, perhaps with a capacity of 800 tHM/a, will be set up around 2020. Recovered uranium and plutonium from reprocessing will go to a demonstration plant and be manufactured into MOX fuel for FBR and PWR. The defence radwaste from reprocessing is at present being conditioned into the proper forms and will be disposed in appropriate repositories. All expertise and experience gained from these practices will be utilized in the future civil radwaste management.

## 1. INTRODUCTION

In the early 1970s, China devoted efforts to establish a complete nuclear fuel cycle system for defence purpose, including the reprocessing of irradiated fuel from plutonium production reactors and reprocessed uranium recycle. Since 1979, China's nuclear industry has changed its emphasis on serving the national economy. Consequently, its nuclear fuel cycle industry is also adjusted to this aim.

In the mid 1980s, it was decided that China should develop nuclear power reactors of the PWR type, meanwhile a closed fuel cycle strategy was also formulated and declared at an IAEA's conference [1]. Therefore, China's spent fuel treatment as the most important sector of the back-end of the nuclear fuel cycle, would involve at-reactor storage, transport, away-from-reactor storage and reprocessing, as well as radwaste management.

## 2. NATIONAL ADMINISTRATION

### 2.1. Management system

China National Nuclear Corporation (CNNC), formerly called the Second Ministry of Machinery Building founded in 1958 and then the Ministry of Nuclear Industry from 1982 to 1988, has been responsible for administration and operation of the whole nuclear fuel cycle. However, since April 1998, its official function has been transferred to the newly composed Commission of Science, Technology and Industry for National Defence (COSTIND), involving the China Atomic Energy Authority (CAEA) for the foreign nuclear affairs. In addition, on behalf of the government, the National Environment Protection Administration (NEPA) and the National Nuclear Safety Administration (NNSA), recently incorporated into NEPA, are in charge of monitoring and administration in the respective fields.

Under the CNNC, the Bureau of Nuclear Fuel and its enterprises implement all operations on spent fuel treatment. The China Institute of Atomic Energy (CIAE) and some universities, such as Tsing-Hua University and Fudan University etc., are involved in most research topics, while the Beijing Institute of Nuclear Engineering (BINE) undertakes design and partly R&D of radiochemical facilities and processes.



## 2.2. Policy and regulation

In order to match the nuclear power growth and to reduce foreign dependency, China is determined to develop a domestic nuclear power and nuclear fuel cycle industry as an essential principle. So far, it has been implemented sound and vigorously.

For PWR spent fuel, the reason why a closed fuel cycle strategy has been made is the following:

- fully utilization of nuclear resources and development of FBR;
- reducing costs of mining, milling and enrichment of uranium;
- disposal of radwaste safely after essentially separation of uranium and plutonium from spent fuel.

Development of civil reprocessing in China is in an advantageous position. The country has a vast land and there are a few ideal sites for locating a reprocessing plant, where the population density is extremely low, and the meteorological and geographical conditions are suitable to this process. In addition, over the past 30 years China has had technical foundation to a certain extent in the field.

A state regulation on spent fuel treatment was drafted by CNNC last year. It is desirable that the regulation could be submitted to the relative government authorities to be reviewed, approved and issued as soon as possible. Unfortunately, there is no formal national long-term development programme for the back-end of the nuclear fuel cycle.

## 3. SPENT FUEL ARISING

China's first domestic nuclear power project, a prototype PWR unit with a capacity of 300 MWe (phase one of the Qinshan NPP in the Zhejiang province), came in operation at the end of 1991. Successively, two PWR units ( $2 \times 900$  MWe) of the Daya Bay NPP in Guangdong province were connected to the electrical grid in 1993 and 1994, respectively. Annually, 60 tHM spent fuel is discharged from these operating units. This number will stay constant until 2002. As of July 1998, the cumulative arisings are about 300 tHM.

The present five-year National Economy Development Plan (1996-2000), contains a total of 8 nuclear power units, which are included in Qinshan phase two ( $2 \times 600$  MWe PWRs) and Qinshan phase three ( $2 \times 700$  MWe CANDUs) imported from Canada, Ling-ao ( $2 \times 900$  MWe PWRs), and Lianyungang ( $2 \times 1000$  MWe WWERs) from Russia, totalling up to 6 600 MWe. Their construction has started and they gradually will be put into commercial operation as of early next century. Therefore, by 2005 the annual PWR spent fuel arisings will sharply increase to 168 tHM, while the accumulative arisings are expected to be 940 tHM. In the meantime, 176 tHM will be discharged annually from CANDU reactors and the cumulative arisings will reach 440 tHM (see Table I).

Several additional large or medium sized NPP projects are being prepared. It is anticipated, that the total installed nuclear capacity, consisting of almost only PWRs with exception of a few CANDU reactors, will reach 20 GWe by 2010 and 40 GWe and 2020. Accordingly, with a constant number of CANDU reactors, the annual spent fuel arisings will reach approximately 600 tHM in 2010 and 1 000 tHM in 2020, while the cumulative arisings will amount to about 3 800 and 12 300 tHM, respectively.

## 4. AT-REACTOR STORAGE OF SPENT FUEL

There is always a storage pool in any reactor building for storage of spent fuel and unloading the whole irradiated core fuel in case of emergency. Spent fuel discharged from reactors has to be stored at the reactor pool for at least 5 years in order to reduce its radioactivity significantly and to simplify the following transport and reprocessing process.

TABLE I. SPENT FUEL ARISING FROM CHINA'S NPPs  
(OPERATING AND UNDER CONSTRUCTION)

Year	# of Units	Installed capacity MWe	Total capacity MWe	PWR/WWER tHM		CANDU tHM		Total tHM
				ann.	cum.	ann.	cum.	
1992	1	300	300	0	0			0
1993	2	900	1 200	12	12			12
1994	3	900	2 100	36	48			48
1995	3	0	2 100	60	108			108
1996	3	0	2 100	60	168	0	0	168
1997	3	0	2 100	60	228	0	0	228
1998	3	0	2 100	60	288	0	0	288
1999	3	0	2 100	60	348	0	0	348
2000	3	0	2 100	60	408	0	0	408
2001	3	0	2 100	60	468	0	0	468
2002	6	2 300	4 400	60	528	0	0	528
2003	9	2 300	6 700	102	630	88	88	718
2004	10	1 000	7 700	144	774	176	264	1 038
2005	11	1 000	8 700	168	942	176	440	1 382

Note: ann.- annual; cum. - cumulative.

However, the at-reactor storage period could likely be extended to 10 years, because most reactor pools can hold spent fuel for this period of time with a compact storage pattern. Reason for increasing the storage period could be that the reactor owners would like to put off delivery of spent fuel to the reprocessor in order to postpone payment, or that spent fuel could not be received by the reprocessor at the appropriate point in time. In fact, the period of at-reactor storage of spent fuel would last maximum 10 years even with a compact storage pattern in all existing and planned NPPs, except for two reactor pools in the phase one of the Qinshan plant which have a capacity of 15 years' fuel discharge.

## 5. TRANSPORT OF SPENT FUEL

China's NPPs are mostly situated on the south and east coastal area while the present reprocessing establishment is in the northwest, which is 3 000 to 4 000 km away from the NPPs. Naturally, the issue of spent fuel transport has to be dealt with.

A feasibility study on the transport of spent fuel from the Daya Bay plant has been completed. The study results have shown that due to lack of rail access to the plant, a combined transport option by both sea and rail would be preferable, using big payload casks and making two round trips annually. Alternatively, a gate-to-gate transport option by road is a realistic solution due to the huge investment of the above option and the very limited business in the near future. However, it is necessary that a completed spent fuel transport system, including casks and its maintenance facility, a purpose-built marine terminal, ships, wagons etc., would be set up for the long term.

### **5.1. Transport cask**

A small-sized cask has been developed, with a 5 t weight and complying with all the regulations laid down by IAEA and accommodating 12 CIAE's Research Reactor fuel assemblies or 7 MTR fuel assemblies. A total of 15 such casks have been transported by road from CIAE to the Lanzhou Nuclear Fuel Complex (LNFC) in two movements. Testing of other type of casks, made of nodular cast iron, with some 20 t weight will be finished next year. Selection of large-sized casks loading more than 10 or even 20 PWR fuel assemblies is being considered to reduce transport unit cost. Purchase or leasing from overseas would be a practical approach for the next years.

Meanwhile, a purpose built test facility, set up in Dalian city, Liaoning province, is capable of handling casks, weighing up to 50 t, and implementing various monitoring and tests, covering shielding, containment, drop, penetration and fire etc.

### **5.2. Transport by sea**

There is a dock without fixed crane in the Daya Bay Plant bordered on the sea. It is not necessary to fit a costly gantry crane with a heavy capacity because of rarely cask handling. Consequently, the operation has to be carried out by a self-provided crane on a specialized ship or a rental floating crane. No doubt, a purpose-built ship with dual hull and bottom is more reasonable and safer than leasing a common ship. A preliminary idea on the former enabling to hold up to 10 cask packages, with a fixed crane and with 2 500 GRT has been projected.

Lanshan Port in the Shangdong province, located at the mid-east coast of China has been screened out for cask transit. A purpose built marine terminal at the port will have a warfare for the ship docking and be equipped with a rail-mounted cantilever crane with a 150 t capacity enabling to load casks from a ship unto specialized wagons or vice versa. A rail track will link the terminal with the main rail line.

The sea route of about 1 130 nautical miles away from the Daya Bay plant has been identified. The navigation conditions have been carefully investigated, including suitable sailing season, haven choice, and prevention of striking on the rocks, as well as the emergency response and salvage access in the event of a ship sinking.

### **5.3. Transport by rail**

Type B(U) package for spent fuel will be transferred to concave shaped wagons with twelve axles and a 150 t loading capacity. China possesses of the capability of fabricating the wagons. After being marshalled and short staying, a special train composed of cask loaded wagons and some necessary auxiliari cars will be driven according to a designated scheme.

A running route between the marine terminal and LNFC has been preliminary selected on the principle of avoiding big and medium sized cities, densely populated regions and the rail routes with the heavy traffic as far as possible. The transport distance is more than 2 600 km, while it would take about a week for a single journey.

### **5.4. Transport by road**

Road transport has the following advantages: flexible, more choices of routes, little restrictions in term of time, geography and environment, easy to avoid some sensitive areas and rush hours, and conveniently organization, in particular without complicated marshalling and transshipment, and less investment. The Chinese road traffic net has been improved significantly. Most roads have reached higher standards. In addition, there are a number of specialized carriers with advanced tractors and trailers having many years of experience in dealing with over-sized and over-weight goods by competent professionals.

The study has shown that transport of spent fuel from Daya Bay Plant to the LNFC by road is technically reasonable and economically acceptable, using super duty tractors and flat trailers imported from abroad with advanced performances and driving the approximate 4 000 km distance, which is dominated by national trunk roads with class two or above standards.

## 6. AWAY-FROM-REACTOR RECEIPT AND STORAGE OF SPENT FUEL

In May 1994, the construction of the first stage of a Centralized Wet Storage Facility (CWSF) started. The facility will have a capacity of 550 tHM (500 t for PWR fuel and 50 t for other fuel) and is located at the LNFC. Beside the storage pools, the facility also covers a receipt and monitoring hall for casks with a overhead crane of 130 t capacity, cooling and purification systems for pool water and the auxiliary shops, such as water make-up, power supply and ventilation etc. [2].

At present, the CWSF's civil engineering has been finished while welding of the stainless steel lining of the pools and installation of equipment are being carried out. It is anticipated, that the CWSF would be put into active operation in 2000, and extended with an additional capacity of 500 tHM or more early next century. Then, it could receive and accommodate at least all the spent fuel from the Daya Bay NPP over its 20 years' joint venture period. In the long term, the facility's storage capacity might again be modularly expanded and inter-linked with the future commercial reprocessing plant through a designated channel.

## 7. SPENT FUEL REPROCESSING

Since the mid 1970s, R&D on spent fuel reprocessing from power reactors has been performed at laboratories of several institutes [3]. In the early 1980s, a multi-purposes reprocessing pilot plant (RPP) project was incorporated in the national economy plan [4, 5]. The project has the following functions:

- demonstration of the processes, equipment and instrumentation under hot conditions;
- experience accumulation of the design, construction and operation;
- training of the operation personnel;
- recovery of highly enriched uranium (HEU) from the High Flux Engineering Test Reactor (HFETR) spent fuel;
- R&D of future reprocessing technologies for LWR-MOX or FBR spent fuel.

The RPP consists of the CWSF mentioned above, a main reprocessing facility (MRF) with a maximum throughput of 400 kgLEU/d, a hot cell laboratory (HCL) with a 900 gHEU/d capacity and a machinery testing workshop (MTW), as well as other auxiliary facilities [3]. The MRF will use a modified Purex process and the HCL a diluted TBP process.

Construction of test rigs at the MTW started in May 1991 and was completed in advance in April 1993. The rigs are designated for simulated tests of some key equipment, instrumentation and remote operation, such as a fuel bundle shear, a set of pulsed sieve extraction columns and their monitoring instruments, control-computed system, remote welding and cutting devices and tele-manipulators etc. Currently, with the exception of the MTW, all RPP buildings, of which construction of the latest facility (MRF) was launched in October 1997, are also actively underway. Hot commissioning of the whole RPP is anticipated by the turn of the century.

After obtaining extensive experience with the RPP and a sufficient amount of accumulated spent fuel, a large-scale commercial reprocessing plant, possibly with a 800 tHM/a capacity, could be commissioned around 2020 in order to match with the nuclear power capacity at that time.

## 8. REPU AND CIVIL PLUTONIUM RECYCLE

In the future, the reprocessed uranium (RepU) as trioxide will be re-enriched and recycled. In the past years, all RepU products, as ammonium uranyl carbonate (AUC), recovered from military reprocessing was converted and re-enriched.

China has been paying attention to R&D on FBR technology. It would possibly become the second generation of nuclear energy in the future. An experimental fast reactor project, with a capacity of 65 MWt and located at CIAE, has been included in the State High Technology Programme and will be completed early next century. Therefore, civil plutonium recovered from reprocessing could be supplied to FBR and/or PWR as MOX fuel. It is currently been considered to build a MOX fuel demonstration facility at the appropriate time.

## 9. RADWASTE MANAGEMENT

The various types of radioactive liquid waste from military reprocessing have been safely stored in mild-steel or stainless steel tanks since the 1970s, while R&D on waste conditioning was initiated in time. All expertise and experience gained from treatment and disposal of defence waste will be utilized in the management of radwaste from civil reprocessing.

### 9.1. Low and intermediate level waste

For low level liquid waste (LLLW) of LNFC in a quite dry climate environment, three open basins have been run since late 1960s while LLLW of the other plant, located in the moist region, is concentrated by evaporation, from which the concentrated LLLW is immobilized by the bituminization process [6]. Since the end of 1992, an engineering facility with twin lines and a throughput of 250 l/h each has been put in active operation.

Both the underground hydrofracture pouring cementation process and the in-situ bulk grouting process for conditioning and disposal of intermediate level liquid waste (ILLW) have been successively developed. A facility with the former process has been operated for 6 campaigns at an appropriate site while another facility with the latter process will complete construction at the LNFC next year. Also, an incineration pilot facility has been tested for volume reduction of spent organic solvent by the pyrolysis process.

Phase one of a Northwest Repository project for solid LLW/ILW, situated at the LNFC site and composed of a set of shallow land burial trenches with 20 000 m<sup>3</sup> capacity, was finished early 1998 and will receive very soon conditioned waste. In the future, it can be expanded to 60 000 m<sup>3</sup> and even to 200 000 m<sup>3</sup> capacity.

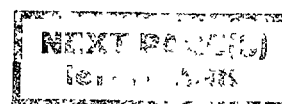
### 9.2. High level waste

A high level liquid waste (HLLW) vitrification route, like the Pamela process in Mol, Belgium, has been defined, using a liquid-fed ceramic melter. Imported from Germany, a non-active full-scale mock-up facility will be set up this year. An active plant is expected to be in operation by early next century.

Finally, vitrified waste accommodated in canisters would be stored in an interim store for at least 30 years prior to ultimate disposal in a deep geological formation, on which a specific R&D programme has started for more than 10 years. Commissioning of a HLW repository is planned around 2050 at a site to be selected by 2030.

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#### **Abstract**

France decided to close the nuclear fuel cycle from the start of its nuclear power programme. The actual strategy is guided by the Waste Act of 30 December 1991. Spent fuel is reprocessed and recycled in 900 MWe PWRs. The La Hague reprocessing plants are performing well. They already reprocessed more than 12 000 t of LWR spent fuel. MOX fabrication plants are fully operational, delivering fuel assemblies for French and foreign utilities. Waste minimisation is a target for the future. The experience already gained in this domain will allow shortly to reduce the volume of HLW and ILW, resulting from reprocessing, to 0.5 m<sup>3</sup>/tU. Other longer term targets are basic knowledge and use of MOX fuel and waste management. This paper describes the current achievements and near-term trends of the 1997 policy, the present status of the French fuel cycle facilities involved and gives an outlook for future evolution.

### **1. THE FRENCH POLICY FOR THE BACK-END OF THE FUEL CYCLE**

The strategic choice of the closed nuclear fuel cycle was decided at the start of the French nuclear power programme. This choice was based on the following criteria: a limited quantity of domestic uranium ore and a strong will to increase the French energy independence.

This goal has been achieved. Energy savings and nuclear power allowed:

- to supply more than three quarter of the French electricity demand with nuclear reactors over the past 10 years (376 TWh nuclear in 1997);
- to increase the France's energy independence from 22% in 1973 to 50% today (while decreasing the CO<sub>2</sub> emissions from about 450 to 350 million tonnes).

In 1991, taking into account the fact that a final repository of HLW was not selected, the French Parliament adopted a law (Waste Act of 30 December 1991), which targeted the year 2006 to define a national policy concerning the future of high-level and long-lived radioactive waste.

This law stipulates that, before 2006, the following activities have to be performed in order to let the Parliament choose the best nuclear policy:

- research on the separation and transmutation of long-lived radioactive nuclides;
- studies for retrievable or non-retrievable underground repositories; and
- studies for waste conditioning and long-term storage.

In 1997, the French Government confirmed the French nuclear policy. The basic guidelines of this policy are:

- a key role of nuclear energy generation;
- reprocessing/recycling for national and international utilities,
- MOX programme implementation of Electricité de France (EdF), and
- the shut down of Superphénix (1250 MWe FBR) and the restart of Phénix (250 MWe), which were part of the recycling scheme.

Subsequently, the main partners adapted their strategies to this new deal:

- EdF, the utility;
- COGEMA, (COmpagnie GÉNÉrale des MATières nucléaires) the nuclear fuel cycle company in charge of reprocessing/recycling;
- ANDRA, (Agence Nationale pour la gestion des Déchets RADIOactifs) the agency in charge of waste repositories, and
- CEA, (Commissariat à l'Énergie Atomique) in charge of research.

EdF, in order to give the nuclear industry the flexibility to adapt to any kind of decision taken in 2006 affecting its activities, has implemented the following strategy (Figure 1 shows the different steps of the nuclear fuel cycle adapted to this strategy and also the involved fluxes and lead times):

- recycling of separated plutonium in twenty eight 900 MWe PWR reactors;
- adaptation of the reprocessed quantities of  $\text{UO}_2$  spent fuel to open recycling possibilities;
- maximum extraction of plutonium during reprocessing and provisional storage of the excess used fuel for future treatment;
- minimisation of final waste volume.

The determining factor to calibrate the fuel cycle is the capacity to recycle MOX fuel in PWRs. This induces MOX fabrication requirements, then reprocessing and interim spent fuel storage requirements and also waste production.

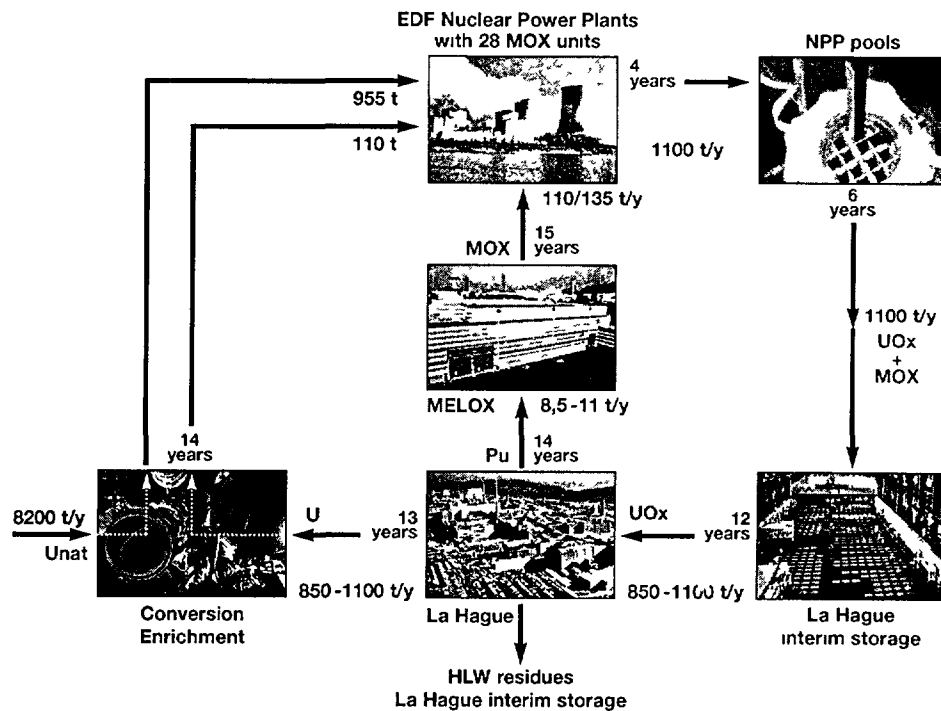


FIG. 1 Fuel cycle back-end in France - reference fluxes 2000-2010

## 2. RECYCLING IN PWRs

French experience in MOX fuel utilisation started in 1974, at the Chooz PWR. It is now rapidly growing. Among the sixteen 900 MWe PWRs authorised (at the end of 1997) to load MOX fuel and starting with Saint-Laurent B1 unit in 1987, 15 reactors have been loaded with MOX. These are: Blayais 1&2, Dampierre 1-3, Gravelines 1-4, Tricastin 1-4 and Saint-Laurent B1&B2 (see Figure 2). At the end of 1997, the recycling totalled 59 reloads and 736 assemblies.

Ten years of using MOX fuel has shown no difference with  $\text{UO}_2$  fuel. Consequently EdF has applied for authorisations to load MOX fuel into 12 additional 900 MWe PWRs. Applications are filled and new authorisations are on their way. Chinon B1 to B4 have just been authorised to load MOX by the end of July 1998.



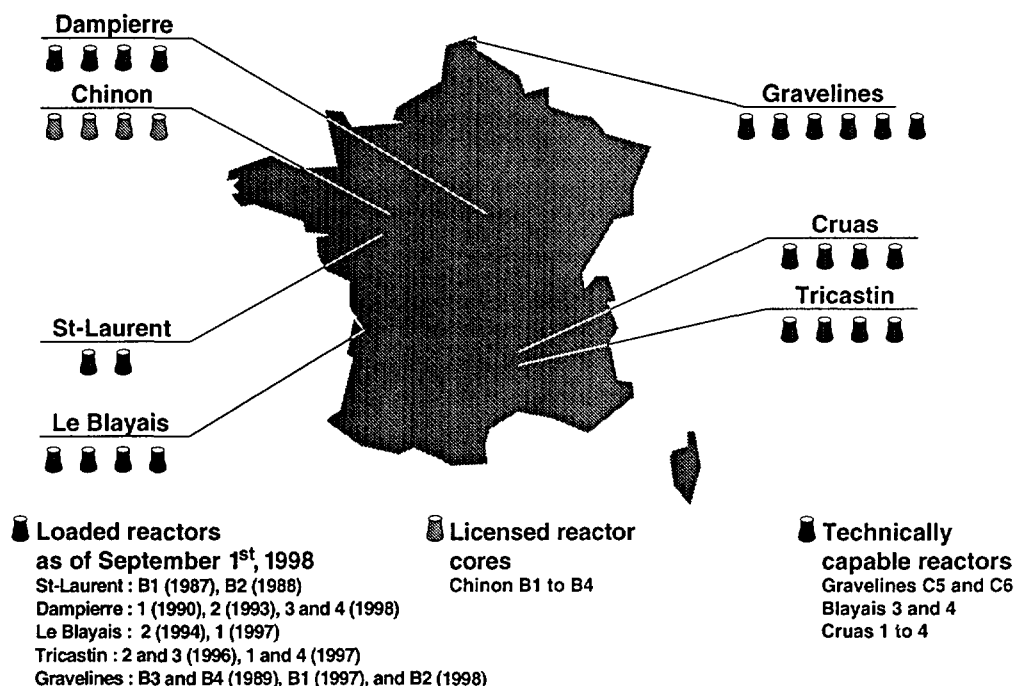


FIG. 2. EdF programme for loading 900 MWe PWR with MOX fuel

Today, according to the granted licence, the MOX content in the reactor core is limited to 30%. Assemblies replacement is performed on a 3 cycle basis for the MOX assemblies and on a 4 cycle basis for the  $\text{UO}_2$  assemblies. For each cycle 16 assemblies of MOX fuel containing 5.3 % of plutonium and 28  $\text{UO}_2$  fuel assemblies containing 3.7% of U-235 are loaded in the reactor. The cycle lasts 12 months (280 effective full power days). The average burnup is 36 GW·d/t for MOX fuel and 44 GW·d/t for  $\text{UO}_2$  fuel. It is the so-called hybrid management.

Table I summarises the different core management strategies which have been or will be applied in the 900 MWe PWR loaded with MOX fuel. A load following mode (quick adjustment of power output to the demand) has been tested at the Saint-Laurent reactors. It is now implemented in all "moxified" reactors.

The French Government maintains a clear and consistent position concerning reprocessing: the Safety Authority requires that EdF only loads fuel assemblies that can be reprocessed. Applications to

TABLE I. CORE MANAGEMENT OF THE 900 MW PWR LOADED WITH MOX FUEL

	$\text{UO}_2$ fuel		MOX fuel		Reload
	Number of cycles	Burnup GW·d/tU	Number of cycles	Burnup GW·d/tHM	
Initial management (from 1987 to 1993)	3	35	3	36	16 MOX and 36 $\text{UO}_2$ (U235 = 3.25%)
Hybrid management (from 1993)	4	44	3	36	16 MOX and 28 $\text{UO}_2$ (U235 = 3.70%)
Parity management (from 2004)	4	44	4	44	12 MOX and 28 $\text{UO}_2$ (U235 = 3.70%)

increase the plutonium content in MOX fuel up to 7% have been submitted by EdF in 1996 in order to recycle plutonium coming from UO<sub>2</sub> fuel with higher burnup. In the mid term, EdF's strategy implies annual supplies of MOX fuel ranging from 110 to 135 tHM and increases in MOX fuel burnup in order to get the same output from MOX fuel as from uranium fuel.

Studies are currently being performed in which MOX fuel assemblies are loaded for 4 cycles. It is the so called "parity" management. In this case MOX fuel and UO<sub>2</sub> fuel have the same energetic value. The plutonium content however has to go up to 8.6%. This type of core management will be used around 2004 and has been tested at Gravelines 4 in 1993, when 4 assemblies were loaded for a fourth cycle and reached a burnup of 44 GW·d/t and in 1996, when another assembly reached a burnup of 47 GW·d/t after a fourth cycle.

Reprocessed uranium (RepU) is recycled as well. The feasibility of this recycling was demonstrated more than ten years ago, in 1987, when eight enriched reprocessed uranium (ERU) assemblies were loaded at the Cruas 4 unit. One assembly reached a burnup of 42 GW·d/t, after a fourth cycle. In 1994, 24 ERU assemblies were loaded into the same unit and, since 1995, two units on this site are reloaded with 40 ERU assemblies each per year. The current enrichment is 3.7%, but it is planned from 1999 onwards, to increase the ERU enrichment up to 4% to obtain the same energy equivalence to classical UO<sub>2</sub> fuel enriched at 3.7%.

### 3. MOX FUEL FABRICATION

In France, plutonium recycling through MOX fuel has been relying upon three fabrication plants: the Belgonucléaire P0 plant at Dessel (Belgium), the COGEMA Cadarache plant and the large size MELOX facility at Marcoule, which started operation in 1995 and is operated by COGEMA. All plants are based on the advanced MIMAS process.

#### 3.1. The Belgonucléaire P0 plant

The Belgonucléaire P0 plant at Dessel (Belgium) started in 1973 with a capacity of 17 tHM/yr. The plant has the largest production record of MOX fuel for LWRs. The nominal capacity is 35 tons HM/y, which has been reached continuously since 1989. The P0 plant has produced a total of more than 300 tHM of MOX fuel. The capacity is to be increased to 40 tHM/y.

#### 3.2. The COGEMA Cadarache plant

The COGEMA Cadarache plant has been processing plutonium fuel for over 30 years, mainly for fast breeder reactors. The plant has the largest experience for plutonium handling. Since 1989, the Superphénix line was converted to LWR MOX fuel fabrication. The production capacity has been progressively increased from 15 tHM in 1993 to 35 tHM today. Further evolution and modernisation will lead to an increased capacity up to 40 t/y. Cadarache is the facility which processed the largest quantity of plutonium in the world (more than 35 tons). In 1997, it delivered 17,700 assembly rods to German utilities.

#### 3.3. The MELOX plant

In 1985, COGEMA decided to launch design studies for the construction of a new large size facility, the MELOX plant, located at the site of Marcoule in the south of France. The Safety Authorities licensed the plant as a Basic Nuclear Facility in May 1990 and authorised start-up with PuO<sub>2</sub> in August 1994. This allowed the fabrication of fuel assemblies from rods delivered by the Cadarache plant. In February 1995, the authorisation for introducing Pu oxide powders was received, and the operation of the entire plant was possible. It is now operating at the full licensed capacity.

The plant has fabricated 100 tons of MOX in 1997. The monthly production record occurred in March 1998, when 19.1 tons have been fabricated. The MELOX capabilities will continue to be

extended in the coming years, see Table II. Applications have already been submitted to increase the licensed capacity to 160 t per year. In 1999, the start up of the West Fitting Building will allow the plant to produce both PWR and BWR multi-design fuel. Japanese utilities will use these new fabrication lines.

TABLE II. MELOX PLANT EVOLUTION

Year	Fabrication technical capacities	Fuel design	Number of fabrication lines	Number of cladding lines	Number of assembling lines
1998	120 t HM	PWR	3	2	1
1999	160 t HM	PWR & BWR	4	2 + 1 multi-design line	1 + 1 multi-design line
>2000	250 t HM	PWR & BWR	4	2 + 2 multi-design lines	1 + 2 multi-design lines

The large scale manufacturing plant meets severe constraints such as using aged plutonium coming from high burnup fuel, high plutonium contents, plutonium isotopic homogenisation and minimal radiation levels for operating personnel. COGEMA might increase the output to 250 tonnes in the next century.

The following specifications have been implemented in order to fit new trends in fuel management and to offer utilities flexible use of MOX fuel:

- Pu-240/Pu total  $\geq 17$  %;
- Pu total/U+Pu+Am  $\leq 12.5$  %;
- Am-241/Pu+Am  $\leq 30\,000$  ppm;
- Average thermal power  $\leq 17.6$  W/kg PuO<sub>2</sub>

These constraints are consistent with the use of plutonium separated from UO<sub>2</sub> spent fuel reaching a 45 000 MW·d/t burnup up and a 6 year ageing. For higher burnup, plutonium should be used earlier after reprocessing. Assuming a plutonium quality such as arising from current reprocessing, the fabrication of MOX fuel dedicated to a burnup reaching 55 000 MW·d/t meets the above constraints as well.

### 3.4 Industrial basis

The significant quantities already produced by the existing facilities and the important capacity coming now under operation, as shown in Table III, give evidence that the MOX fuel fabrication has reached its industrial phase. EdF and other foreign utilities will be provided with important plutonium recycling possibilities on a well-mastered technological and economical basis.

TABLE III. INDUSTRIAL MOX FABRICATION PLANTS CAPACITIES

Country	Company	Location	1998 capacity	Capacity next century	Pu consumption next century
Belgium	Belgonucléaire	Dessel P0	35 t	40 t	2 to 3 t/a
France	COGEMA	Cadarache	35 t	40 t	2 to 3 t/a
	MELOX	Marcoule	120 t	250 t	10 to 12 t/a

## 4. REPROCESSING

Along with the French nuclear programme, the La Hague reprocessing plants (UP2 and UP3) are serving utilities from several European countries (Germany, Switzerland, Netherlands and Belgium), as well as Japanese utilities. They are operated by COGEMA and the French LWR spent fuel undergoes reprocessing in the UP2 plant, whilst the UP3 plant is devoted to reprocess 7 000 tons

of spent fuel coming from foreign European and Japanese utilities until the year 2000. The present section describes the innovative features of the La Hague plants, as well as the operational experience already acquired.

#### **4.1. History and status**

The UP2 plant started in 1966 and has reprocessed various types of fuel, however since 1987, the plant is devoted to reprocess LWR fuel only. To meet France's nuclear power programme requirements, the annual capacity of UP2 was increased from 400 to 800 t by adding new facilities (UP2-800) in the early nineties. The total production of UP2 is 7 246 t as of 1 April 1998.

It should be noticed that industrial demonstration of reprocessing FBR spent fuel (9.9 t from 1979 to 1984) and MOX spent fuel (9.4 t in 1992 and early 1998) was performed at La Hague plant, without any kind of difficulty.

Three decades experience at La Hague demonstrates the industrial mastery of commercial reprocessing. The whole reprocessing-conditioning-recycling production line (La Hague, Cadarache and Marcoule) has been certified ISO 9002 in 1997.

With the UP2 and UP3 plants, which have been operated at full nominal capacities for the past 4 years, this unique industrial complex is able to serve nearly 100 PWRs offering high quality recyclable energy products and conditioned residues.

For the coming years, 850 tonnes of EdF's spent UO<sub>2</sub> fuel will be reprocessed "on line" each year. Excess quantities of unloaded spent fuel will be stored awaiting later treatment in order to maintain the parity between recycling possibilities and reprocessed quantities. UO<sub>2</sub> fuel will be reprocessed first.

#### **4.2. Main innovative features of UP3 and UP2**

Large investments in the La Hague complex are almost completed. Only two new workshops, which will make the La Hague complex more homogeneous and still more effective, will come on line in year 2000: the R4 and the ACC workshops. R4 is involved in the separation-purification process. It will save one purification cycle by replacing the MAU and MAPu workshops.

ACC is dedicated to the hulls and end-fittings compaction before conditioning in universal canisters. In the old process, hulls and end-fittings were conditioned in drums of concrete. The new process will decrease 5 fold the volume of waste. It will also simplify overall waste management as only one type of canister will remain: the Universal Canister. This volume reduction is shown on Figure 3. This new technique will be applied from 2000. Until that date, hulls and end-fittings are stored under water awaiting compaction.

#### **4.3. Operational experience**

The reprocessing complex at La Hague reached full capacity in 1995. The acquired experience shows outstanding achievements for operational and process performances as well as for environmental impact in terms of personnel exposure and waste minimization.

##### *4.3.1. Operational and process performances*

The facilities are operating at full capacity with a recovery rate in accordance with the required specifications. The uranium and plutonium recovery rates are 99.88% (Table IV). The plant reliability and availability are well demonstrated by the reprocessed quantities which are given in Figure 4:

- UP2 has reprocessed 7 246 tons of which 2 150 t U for foreign customers;
- UP3 has reprocessed more than 5 042 t U, exclusively for foreign countries.

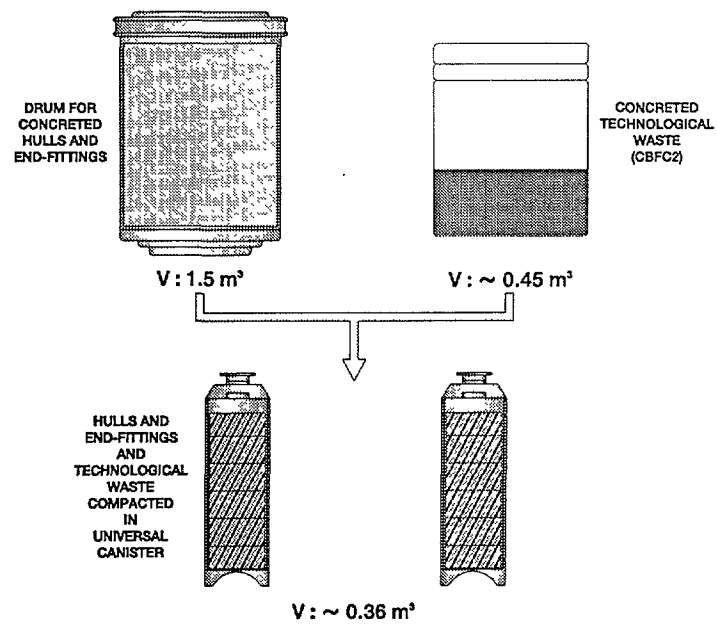


FIG. 3. Plant for compaction of hulls and end-fittings

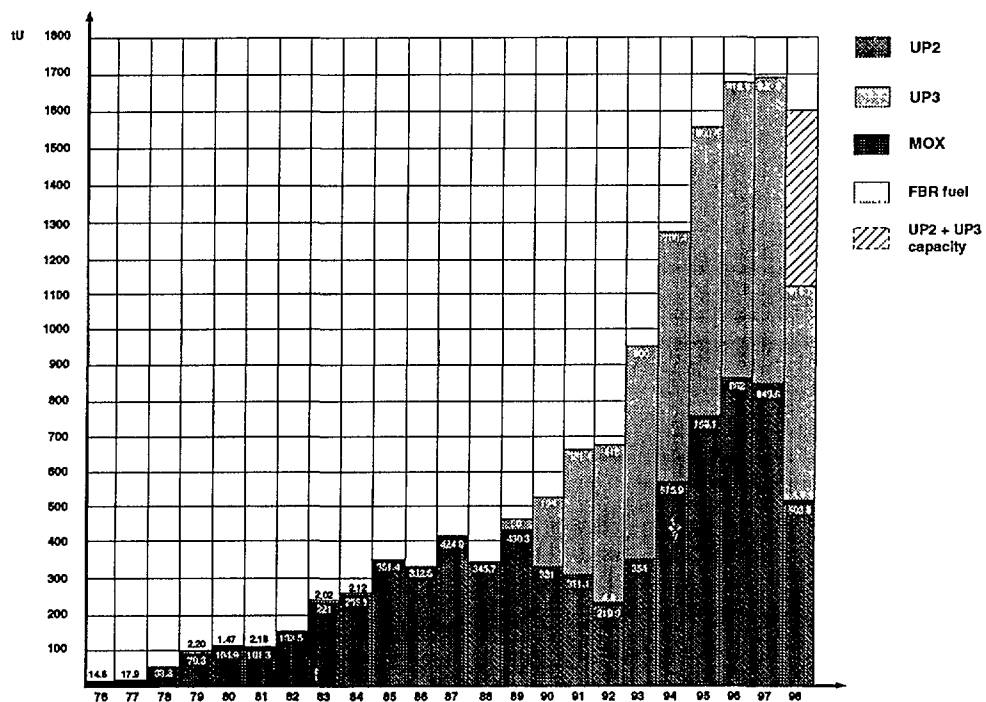


FIG. 4. Annual reprocessed quantity of spent fuel in UP2/UP3, as of August 1st, 1998

TABLE IV. U, PU SEPARATION EFFICIENCY

	Uranium	Plutonium
U, Pu recovered	99.88 %	99.88 %
Glass and other waste	0.12 %	0.12 %
Total	100%	100%

#### 4.3.2. Dosimetry and environmental impact

##### Occupational exposure

While current regulations require the operating personnel exposure to be limited to 50 mSv/y, the UP3 and the UP2 facilities have been designed so that the number of plant workers receiving a 5 mSv/y dose is zero or near zero.

Actual exposure values have been continuously decreasing over the few past years while production increased. The collective radiation dose reached 0.59 man·Sv/y in 1996, for the global exposure while the average occupational exposure for personnel is now reduced to 0.16 mSv/y, which is more than 10 times less than natural radiation exposure. The evolution of these two measures from 1976 to 1996 is shown on Figures 5 and 6.

##### Impact on La Hague site

Liquid and gaseous effluents of La Hague plants are well under the authorized limits (on average ten times lower) and the impact on the surrounding site is very low: man-made radioactivity detected in algae, sand, sediment, seafood, dust, rain, ground and surface water as well as in vegetation and milk is negligible (see Table V).

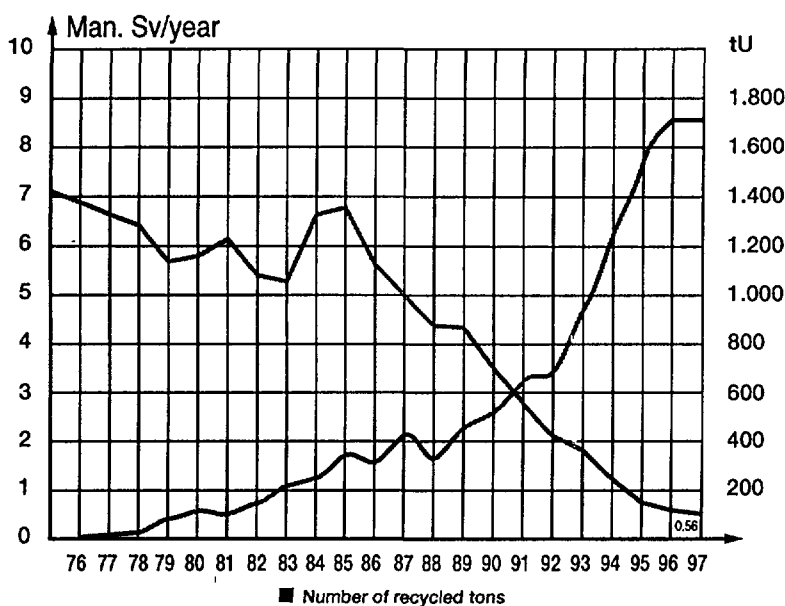


FIG. 5. Average exposure for UP2 + UP3 plants (operation + maintenance)

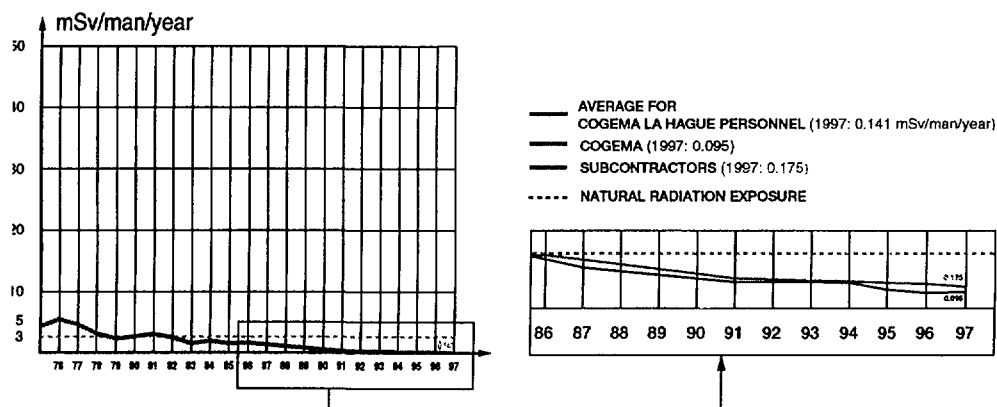


FIG. 6. Average annual exposure of COGEMA/subcontractors personnel monitored at La Hague

#### 4.3.3. Waste management

At the time of commissioning of La Hague, 4 types of waste were forecast to be produced :

- glass, containing the fission products, gathering 99.5 % of the activity of the total waste;
- concrete containers with hulls and end-fittings;
- bitumen drums for sludges;
- grout concrete containers with technological waste.

The excellent process performances achieved at UP3 made it possible to launch a waste minimization programme for UP3 and UP2, mainly based on an improved effluent management and on the use of additional evaporation capacities. It is now possible to route practically all the activity towards the vitrification units. The need for precipitation will thus disappear in normal operation for the low and medium level effluents. The resulting small increment of activity incorporated in the glass will induce no noticeable volume increase. With these improvements and the compaction of the hulls and end-fittings, the overall volume of high level and long-lived waste, resulting from reprocessing, decreases below  $0.5 \text{ m}^3/\text{t}$  (see Figure 7) that is:

- 6 times less than the design values; and
- 4 times less than that of direct disposal (estimation).

## 5. SPENT FUEL INTERIM STORAGE

Sites, where it is impossible to transfer spent fuel to any other storage facility (centralised or at the reprocessor storage sites), and utilities, who want to balance the reprocessing and recycling flows, will have to adapt their spent fuel storage capacities. In France, EdF will transfer all its spent fuel to the La Hague storage pools where it will cool down before on line reprocessing or will be stored for later treatment. The La Hague pools capacity is about 14,000 tons of spent fuel.

Abroad, adaptation of storage capacities has already occurred. In the coming years, additional spent fuel storage requirements will increase. Different technologies may be proposed to create additional storage capacity. Pools re-racking is an effective way to increase the capacity. The technology is well known and commonly used today. The use of dry casks brings additional capacity on site. Spent fuel assemblies are loaded into storage casks or into dual purpose casks which combine the transport and storage functions (such as TN-24, TN-40 and TN-45 of TRANSNUCLÉAIRE). This system offers the best financial modularity as investments are adjusted to the annual storage needs.

**TABLE V - TOTAL NATURAL AND MAN-MADE RADIOACTIVITY  
DETECTED AT THE LA HAGUE SITE**

Medium	Type of sample	Unit of measure	Radioactivity				
			Natural <sup>a</sup>	Reference year	Man-made		
					Past 3 years		
			1996	1965	1994	1995	1996
Air	Dust	mBq/m <sup>3</sup>	8 600	(9.9)	0.8	0.8	0.9
	Rain	Bq/l	0.9	(2.2)	0.2	0.2	0.2
Water	Ground water	Bq/l	1	-	0.8	0.9	0.9
	Surface water <sup>b</sup>	Bq/l	1.1	-	0.7	0.7	0.6
Earth	Vegetation	Bq/kg fresh	293	(210)	0.7	0.7	0.7
	Milk	Bq/l	48	(7.4)	0.4	0.4	0.3
Sea	<b>Algae</b>						
	Fucus	Bq/kg fresh	272	(48)	9	7	6
	Lichen	Bq/kg fresh	200	(100)	10	9	6
	Corallina	Bq/kg fresh	111	(110)	10	8	7
	Laminaria	Bq/kg fresh	291	(15)	4	3	3
	<b>Sand Sediment</b>						
	Sand	Bq/kg fresh	426	-	6	6	5
	Sediment	Bq/kg fresh	358	-	28	20	14
	<b>Seafood<sup>c</sup></b>						
	Crabs	Bq/kg fresh	58	-	4	3	2
	Oysters	Bq/kg fresh	30	-	2	2	1
	Mussels	Bq/kg fresh	33	-	2	2	2
	Limpets	Bq/kg fresh	69	(30)	7	4	3
	Fish	Bq/kg fresh	95	-	2	2	2

<sup>a</sup> Including uranium, thorium and their daughter products, potassium and beryllium which occur naturally in the earth's crust,

<sup>b</sup> No activity was detected in the majority of surface water samples. However trace amounts of radionuclides were detected in the Moulins and Sainte-Hélène streams, but their average activity remained well below the applicable standards,

<sup>c</sup> Analyses are performed on edible portions

Notes Activities in parentheses are from radioactive fallout weapons testing, and not from La Hague

**Measurement thresholds:** beta = 0.2 Bq/l

Radioactivity of human body = 135 Bq/kg

Radioactivity of sea water = 14 Bq/l

Fuel assemblies may also be dry stored in vaults like CASCAD, a French interim storage facility in operation since 1990 (storage of Brennilis and naval reactor fuel). The main feature of such storage facility is a concrete structure with metallic wells in which the fuel assemblies are securely stored. Fuel containment is guaranteed by a double barrier: the canister and the well. The design (done by SGN) may be adapted to any kind of nuclear fuel.

## 6. WASTE MANAGEMENT

Radioactive waste management includes different types of waste. In France, low level waste (LLW) is disposed off at the SOULAINES surface disposal center managed by ANDRA. The intermediate level waste (ILW) and the high level waste (HLW) will be managed according to the Waste Act of December 1991. ANDRA is in charge of recommending sites for the construction of underground laboratories designed to test deep geological repository concepts. The selection of two



sites for underground laboratory construction should be done in the coming months. The final decision is to be taken before 2006 by the French Parliament.

French ILW and HLW conditioned by the La Hague plants are stored on site before their transfer to the future ANDRA site, while foreign residues are or will be sent abroad. Three transports of vitrified waste towards Rokkasho-Mura (Japan) and one to Gorleben (Germany) have already occurred.

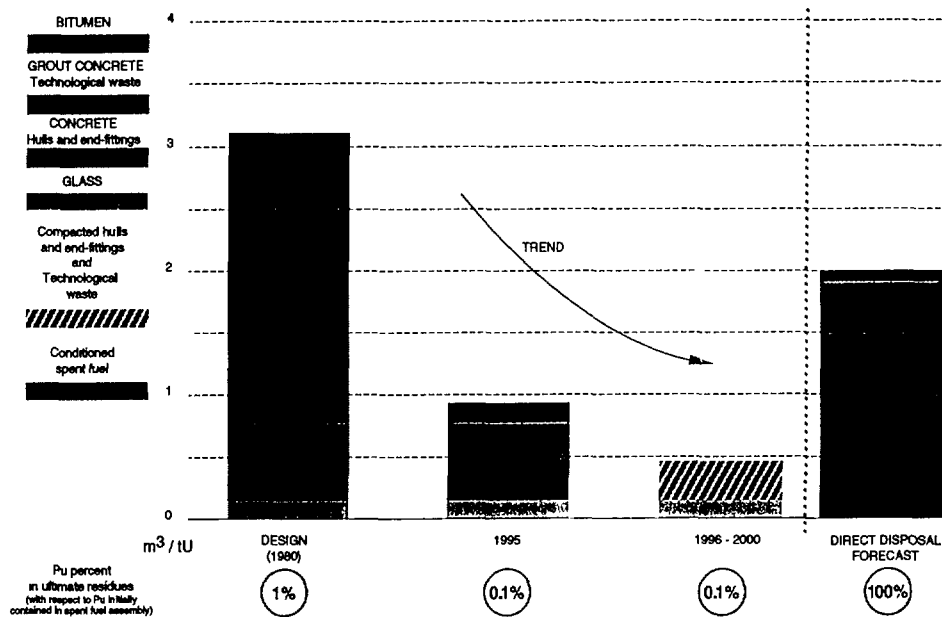


FIG. 7. Volumes of final residues generated in UP3 (long-lived waste after conditioning)

## 7. LONG-TERM TRENDS

The French nuclear industry is firmly committed to the recycling policy. It proves its capability of mastering the entire closed fuel cycle. However, research and development activities are carried out to continuously improve waste minimisation in reprocessing plants and knowledge of MOX fuel behaviour in reactors.

### 7.1. Waste management research

The Waste Act of 30 December 1991 stipulates that before 2006, research on the separation and transmutation of long lived radioactive nuclides in the waste, has to be performed. According to these objectives, the CEA launched a major long-term R&D programme, called the SPIN (SéParation INcineration) programme, on separation and incineration of long-lived nuclides (i.e. actinides). SPIN is divided into different programmes.

The Puretex programme will be put in operation in 2000. It focuses on volume and activity reduction of ILW type waste from reprocessing, i.e. waste with long-lived nuclides content and low heat load, and also on the reduction of activity releases to the air and to the sea. In 1980, the volume of ILW per ton of reprocessed spent fuel was 3 m³, it was 1 m³ in 1995 and will be 0.5 m³ in 2000.

The long-term Actinex programme focuses on separation and transmutation of long-lived nuclides in order to reduce waste toxicity. While separation research investigates process routes and

types of extractants, transmutation research is looking at data validation for incineration tests (e.g. cross-section, assay and decay data) and parametric studies assessing the feasibility of actinide incineration (e.g. reactor type, fuel type). This research is performed by the CEA.

First results show that neptunium separation by the tributylphosphate solvent might be done in existing facilities without important modification. Further investigations are required on technetium separation and the separation processes Diamex and Sanex. The Diamex process allows to separate fission product from a mixture of lanthanides, americium and curium. Its feasibility has been demonstrated. The Sanex process allows to separate lanthanides from americium with curium; however, the process performance has to be improved. Such separations will have to be performed in new, complex and expensive facilities.

When separated, fission products and minor actinides will have to undergo transmutation in a reactor. The 250 MWe fast neutron reactor, Phénix, will be used for such experiments before its closure in 2004.

## **7.2. Improvement of basic knowledge of MOX fuel**

In order to acquire a similar knowledge as for  $\text{UO}_2$  fuel and to validate increased burnup plans, several actions are underway, e.g.:

- fresh product characterisation;
- post-irradiation examination including characterisation with respect to fabrication parameters;
- neutron physics critical experiments;
- analytical programme in experimental reactors;
- theoretical studies of different management schemes.

## **7.3. Future trends in recycling**

MOX assemblies have been designed to be reprocessed in La Hague and the guarantee of their reprocessing was one of the points to be demonstrated to the French Safety Authorities. From 1991, in CEA laboratories and from 1992, in La Hague industrial plant UP2 reprocessing MOX fuel was conducted with quite satisfactory results. Thus, the way is open for obtaining maximum energy from recycled plutonium. Several pathways are possible according to reactor types and design:

### **Recycling in PWRs:**

- A standard PWR reactor generates 30 kgPu/TW·h;
- When fuelled with 30 % MOX as currently implemented, the same reactor gives a near zero plutonium balance;
- Advanced PWR reactors accepting 100 % MOX loading, will burn plutonium at a rate of 60-80 kgPu/TW·h.

### **Recycling in fast reactors:**

- Reactors such as Phénix or Superphénix were initially conceived for a net positive balance of plutonium (breeder reactors);
- The same type of reactors, if the blankets are removed, will consume plutonium at a rate of 20 kgPu/TW·h;
- Advanced fast (burner) reactors could reach high consumption rates up to 100 kgPu/TW·h. The CEA's CAPRA programme is to demonstrate the feasibility of designing and operating such high plutonium burner reactors.

The plutonium balances for various reactors types are shown on Figure 8. As plutonium consumption in fast neutron reactors is limited due to the closure of Superphénix most recycling will be done in PWRs.

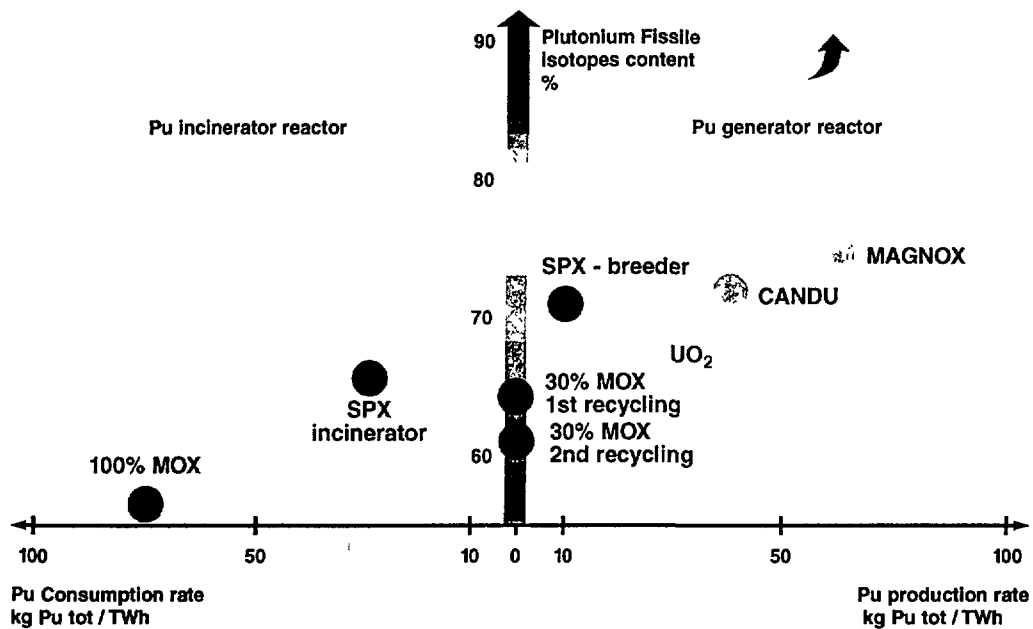


FIG. 8. Plutonium balances for the various reactor types

Although moxification of 1,300 MWe PWRs (P4 & P'4 models) and in 1,450 MWe PWRs (N4 models) is not for the near future, longer term strategy studies include MOX fuel use in these reactors. In this case, the plutonium content in MOX fuel shall reach 9%.

## 8. CONCLUSION

Reprocessing followed by recycling of plutonium and uranium is in France the fundamental strategy for dealing with spent nuclear fuel. Some other countries adopted the same strategy. The present status relies on improved and well-mastered technologies on an industrial scale, and on adapted capacities.

Besides the industrial maturity, future improvements are being prepared steadily. Consistent industrial partnership and relevant R&D programmes will lead to increased performances regarding natural resource management, environment aspects and economic results.

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# STATUS OF POWER REACTOR FUEL REPROCESSING IN INDIA

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## Abstract

Spent fuel reprocessing in India started with the commissioning of the Trombay Plutonium Plant in 1964. This plant was intended for processing spent fuel from the 40 MWth research reactor CIRUS and recovering plutonium required for the research and development activities of the Indian Atomic Energy programme. India's nuclear energy programme aims at the recycle of plutonium in view of the limited national resources of natural uranium and abundant quantities of thorium. This is based on the approach which aims at separating the plutonium from the power reactor spent fuel, use it in the fast reactors to breed  $^{233}\text{U}$  and utilise the  $^{233}\text{U}$  generated to sustain a virtually endless source of power through thorium utilisation. The separated plutonium is also being utilised to fabricate MOX fuel for use in thermal reactors. Spent fuel treatment and extracting plutonium from it makes economic sense and a necessity for the Indian nuclear power programme. This paper describes the status and trends in the Indian programme for the reprocessing of power reactor fuels. The extraction of plutonium can also be seen as a far more positive approach to long-term waste management. The closed cycle approach visualised and pursued by the pioneers in the field is now steadily moving India towards the goal of a sustainable source of power through nuclear energy. The experience in building, operating and refurbishing the reprocessing facilities for uranium and thorium has resulted in acquiring the technological capability for designing, constructing, operating and maintaining reprocessing plants to match India's growing nuclear power programme.

## 1. INTRODUCTION

The need for power generation to meet the growing energy needs of a vast and developing country like India needs no emphasis. In comparison, the energy demand in the advanced countries is nearly saturated. Nuclear power presents itself as a proven technology and a viable alternative to the depleting fossil fuels besides providing a cleaner environment. India has made steady and substantial progress in all aspects of nuclear power generation following a planned strategy to evolve a balanced programme of development of the complete nuclear fuel cycle. Spent fuel reprocessing is one of the vital areas of this programme in which India has achieved considerable success. Closing the nuclear fuel cycle through reprocessing and recycling the plutonium is not only prudent from the point of optimal utilization of our resources but is also a responsible option because reprocessing separates the long-lived actinides from the fission product waste, thus addressing a major long-term safety concern of high level waste disposal.

At present, India has two BWRs and eight PHWRs under operation with a modest installed power capacity of 1840 MWe. India opted for PHWRs as the first stage reactors of its three stage nuclear power programme, taking into account the availability of the natural uranium resources and the industrial infrastructure in the country. Four more units of 220 MWe are in an advanced stage of construction. The design of two units of 500 MWe to be set up at Tarapur is ready, two PWRs units of 1000 MWe each are being set up at Kudankulam, in the South of India as an Indo-Russian joint venture. A Fast Breeder Test Reactor (FBTR) using mixed uranium-plutonium carbide fuel core has been built at Kalpakkam. Development of a 500 MWe Prototype Fast Breeder Reactor (PFBR) is under way. Utilisation of thorium in the research reactors and power reactors for the production of  $^{233}\text{U}$  has been established along with the facilities for its separation.

The planned growth of nuclear power demands a sustained and accelerated reprocessing activity to support this programme and to treat the fuel arising from these reactors.

## 2. REPROCESSING IN INDIA – A PERSPECTIVE

### 2.1. Plutonium plant, Trombay

The beginning of the spent fuel reprocessing activity in India goes back to nearly four decades, when a decision was taken in the late 1950s to build the first plutonium separation plant at Trombay, Mumbai (earlier known as Bombay). When embarked on this front-line technology, India did not have

a strong industrial or R&D infrastructure. The challenge was, however, met when this plant was designed, constructed, tested and commissioned during 1964. The first spent fuel rod from CIRUS was charged into the dissolver by Dr. Homi Bhabha, the visionary in the Indian nuclear history. This plant was operated successfully for a number of years.

The Trombay Plant adopted the Purex flowsheet using mechanically pulsed solvent-extraction columns with 30% tributyl phosphate as solvent. To achieve the desired quality of the products, two cycles of co-decontamination with final purification cycles for uranium and plutonium were adopted. As the fuel was uranium metal with aluminium cladding, the dejacketing was carried out by chemical means. Plutonium was purified by anion exchange and the uranium was subjected to final purification by solvent extraction. The purified plutonium nitrate solution was further converted to oxide or metal as required.

Later, the plant underwent a complete decontamination and decommissioning cycle and was refurbished with enhanced processing capacity incorporating improved features and has operated since 1983 with interruption on two occasions. The direct maintenance concept was adopted for the Trombay Plant and this proved highly successful as, during operation, many parts of the plant could be approached after decontamination as and when required in order to effect modifications to suit operational requirements.

This plant has helped a great deal in generating expertise and trained manpower in the design, execution and operation of future reprocessing plants in the country and has fulfilled all the requirements for which it was built.

## **2.2. Prefre-1, Tarapur**

Meanwhile, a plant to process zircaloy clad uranium oxide fuel was built at Tarapur on the basis of experience at the Trombay Plant and extensive development studies carried out on a pilot scale. Design of this first Indian Power Reactor Fuel Reprocessing Plant (Prefre-1) incorporated features based on the experience gained from the plutonium plant, Trombay. This plant was commissioned during 1974. This plant, like the plant at Trombay, also adopted the Purex process with the direct maintenance concept. The plant has been designed to handle zircaloy clad oxide fuel. The chop-leach method has been chosen for the head-end treatment in preference to the chemical decladding, in view of the excessive corrosion problems expected from the chemicals required and the resulting waste management problems. As against this, the mechanical chopping of the fuel into small pieces before dissolution involves a number of mechanical operations. These operations must be carried out remotely and good performance of the equipment has been experienced. The Prefre plant introduced, for the first time, some of the problems unique to the processing of ceramic fuels of relatively higher burnup. These problems were satisfactorily resolved in course of time by innovations made in the process steps adding to Indian confidence in the R&D strength built-up over the years.

A special feature of this plant was its design meeting the requirements of international safeguards. This plant has so far carried out several reprocessing campaigns with fuels from RAPS and MAPS, the former under IAEA safeguards.

## **2.3. KARP Kalpakkam (Prefre-2)**

This is the second oxide fuel reprocessing plant built to provide a reprocessing base in the southern region of India. This plant has been completed and has undergone extensive cold commissioning trials with natural uranium. The plant is now ready for hot processing. The plant design has taken into account our operation and maintenance experience and has been built with an upgraded technology. The plant also has provision to process the spent fuel from the FBTR in a separate set of cells. A hybrid maintenance concept has been followed in the design of KARP in which the maintenance of failure-prone hardware, such as metering pumps, in the high active and high plutonium circuits will be carried out in hot cells using remote handling equipment, like servo

manipulators. All other process systems in medium and low active cycles will have contact maintenance.

The fuel handling area has a spent fuel storage pool with provision for horizontal storage of fuel, a pool-water purification system and associated mechanical handling equipment like over-head crane, pool bridge with motorized tong assembly unit, etc. It is from this area that the spent fuel is charged through a transfer port embedded in the head-end cell wall into a fuel magazine with an automatic pusher for feeding the fuel to the chopper. The chopped fuel pieces are then dropped into a dissolver through a distributor. In addition, the head-end cell is equipped with an in-cell crane, a pair of master-slave manipulators, shielded viewing windows, closed-circuit television camera, etc., for operation and maintenance. After the dissolution of the fuel, the undissolved hardware like zircaloy is transferred remotely to a drum mounted on a motorized trolley for retrieval through an underground tunnel extending to the fuel-handling area into a shielded cask for suitable disposal as radioactive solid waste.

The solution from the dissolver is moved out for further processing by solvent extraction. The process equipment and piping are installed in seven concrete cells of varying thickness. The first cell houses the conditioners for the feed solution and the co-decontamination-cum-partition cycle with associated evaporators, scrubbers, etc., for separation of the bulk of the fission products. The maximum activity is handled in this cell. Equipment for the uranium purification cycle is installed in the second and third cells. Equipment for the final plutonium purification cycle are housed in the third cell, from which the purified product solutions are taken out into the plutonium re-conversion facility for further processing. In this area, the plutonium nitrate solution is processed in a series of glove-box trains for conversion to plutonium oxide through the continuous oxalate precipitation step. The depleted uranyl nitrate solution from the third cell is moved to a separate area for further purification, if necessary, by passage through a silica gel column and then precipitation to ammonium diuranate, de-watering and calcination to uranium oxide. To ensure safety, the utmost care is taken of ventilation to maintain the desired pressure gradients and airflow patterns. For this purpose, the whole plant is divided into different zones, each identified according to the radiological status of the areas involved. The air from the active areas is exhausted through absolute filters before discharging through a tall stack.

The highly radioactive and intermediate level liquid waste solutions are concentrated and stored in waste storage tanks for an interim period before being sent for immobilization. All low level effluents are suitably treated, monitored and discharged after ensuring that the activity is within the permissible limits stipulated in the radiological health and safety regulations.

The Plant Control Room is located in a separate adjoining building with all the instrumentation connections brought out from the process building. This building also houses the plant room, consisting of the compressed air plant, air-conditioning plant, equipment for the air ventilation system, electrical switch-gear room, etc. Personnel entry into the process building is through the change rooms provided in this building.

India now possesses the expertise which covers various aspects of the reprocessing technology starting with the conceptual plant design to the detailed engineering, fabrication, installation, commissioning, decommissioning and operation of the reprocessing plants handling different types of spent fuel. In the fulfilment of these objective, a strong R&D base has also been built-up in diverse fields encompassing engineering, instrumentation, process chemistry, computer applications, robotics, corrosion studies etc.

Thus, the journey from the plutonium plant, Trombay to Kalpakkam reprocessing plant represents a progressive evolution in the plant design, based on a better understanding of the complexities of the technology. The safety performance of the plants has improved significantly and a reduction in occupational exposures, environmental releases and the waste volume, as a direct

outcome of various design improvements, could be achieved. Experience gained in these plants will be useful in introducing fully integrated remote maintenance features of future plants.

Another encouraging achievement over these years has been the evolving close interaction of the Indian industry with the quality standards demanded by the nuclear sector. When the plutonium plant was constructed at Trombay, almost all the fabrication and installation works were carried out in-house. In the case of KARP, many major fabrication and erection jobs have been carried out by the Indian industry both in the private as well as in the public sectors. This partnership approach is crucial for the success of the Indian programme in the future.

In the context of the projected spent fuel arising from the PHWR programme, need exists for enhancing the reprocessing capacity to create a plutonium base for the fast breeder programme and also for fuelling the thermal reactors with MOX.

### 3. TECHNICAL ASPECTS

#### 3.1. Oxide fuel processing

The well established Purex process has been used in all the reprocessing plants with suitable head-end treatment depending on the fuel type. The aluminium cladding for the metal fuel used in the research reactors is removed by dissolving it in sodium hydroxide. In the case of zircaloy clad power reactor oxide fuel, the chop and leach method has been adopted. This is followed by dissolution in nitric acid, solvent extraction and final processing of plutonium by oxalate precipitation, filtration and calcination.

The main steps in the reprocessing of the oxide fuels are:

- Chopping the fuel bundle by a fuel chopper;
- Dissolving the exposed fuel in hot nitric acid;
- Feed clarification to remove the suspended particles prior to solvent extraction;
- Feed valency adjustment;
- Isolation of major quantities of fission products from uranium and plutonium and the separation of plutonium from uranium;
- Further purification by independent solvent extraction cycles for uranium and plutonium;
- Conversion of U and Pu product solutions to their oxide forms.

The other process steps include the evaporation and acid destruction to reduce the waste volume storage requirements, off-gas treatment for the gaseous effluents, solvent treatment, acid recovery and their recycle etc. Interim waste storage facilities for acidic and alkaline waste have been provided in all the plants before their treatment by way of immobilisation in glass matrix and cementation.

#### 3.2. Fast reactor fuel processing

The Purex flow sheet is time tested and has established itself as a reliable and efficient separation route. Even though there has been a rich experience in the reprocessing of thermal reactor fuel, there is yet to gain experience in the treatment of fast reactor fuel which is a highly challenging area in view of the higher levels of plutonium, irradiation levels and fission products involved. It appears that the Purex flow sheet will be quite adequate even for the fast reactor fuel if centrifugal contactors are utilised to shorten the residence time which can minimise the operational problems associated with the solvent/diluent degradation products.

However, non-aqueous and pyrochemical processes also appear attractive alternatives. These methods provide: (a) shorter doubling time achievable with such fuel, since these processes can withstand higher levels of irradiation and thus can be used with shorter cooled fuels; (b) compact process and waste volume arising in such processes; and (c) actinides present in the fuel can be separated along with the plutonium stream and by virtue of their having favourable fission cross sections can be recycled as fresh fuel along with the plutonium. The long-term storage hazard of the vitrified waste product is also thereby reduced.

### 3.3. Design considerations

High intensity radioactive materials and highly corrosive reagent are encountered in a spent fuel treatment facility. These, along with nuclear criticality considerations impose several safety related requirements in the design, operation and maintainability aspects of the facility. Equipment sizing, spacing, ease of decontamination, access for replacement, nuclear criticality control, meticulous nuclear material accounting and control being some of these considerations. Mechanical systems in the head-end part of the facility are handled by remote maintenance. For rest of the process systems, the direct maintenance philosophy has been adopted and provisions exist for extensive decontamination. Redundancies have been provided by way of duplicate equipment, transfer modes etc.

In order to prevent premature failure of equipment by corrosion, care has been taken to select the proper materials and fabrication techniques. Nitric acid grade stainless steels with very low corrosion rates and special alloys like titanium etc. are being adopted as the material of construction for the process equipment which encounter high temperatures and high nitric acid concentrations.

Currently, uranyl nitrate is being utilised as reductant for Pu partitioning. However, this results in an increase in the processing load on the plant equipment. Development work is in progress to introduce an in-situ electrolytic reduction technique or some other suitable methods in future facilities.

Metering pumps of hydraulically actuated diaphragm type have been used in the reprocessing plants for many years. The most frequent problems with such pumps have been the diaphragm rupture. Use of filters on the pumped streams and modified suction piping have reduced their recurrence. Alternate solution transfer modes utilising the airlift principle and flow-metering devices have been developed.

The plants use perforated plate air pulsed columns as the solvent contactors and the performance of both the columns and the air pulsing systems have been satisfactory. A modified type of mixer-settler contactor has been developed utilising a combined air-lift and static mixing principle which eliminates the need for maintenance prone components.

## 4. DEVELOPMENT ACTIVITIES

A number of development studies are under way in the field of reprocessing with a view to improve the performance of the plant systems in the currently operating plants and also to develop new or modified equipment and processes for use in future plant projects. Studies are being conducted to develop modified solvent contactors, continuous processes in place of batch processes, development of system components like column pulsing, pulsed pumps/fluidic devices, introduction of enhanced automation in the plant and process control to improve the productivity of the existing and future plants. These efforts include development of equipment for fuel chopping, feed clarification, continuous dissolvers for zircaloy clad power reactor uranium/thorium oxide fuel, improved reduction methods of partitioning by electrolytic or other suitable means, robotisation of sampling system etc. Efforts are also going in for the development of flowsheets for routing specific nuclides within the process and to minimise generation of waste streams. Development activities have continued in the synthesis of suitable extractants for actinide separation and to study their effectiveness. Development work on solvent purification for recycle within the plant by vacuum distillation has been taken up. The subsection below describe briefly the extent and scope of the planned development activities.

### 4.1. Head-end systems

This head-end process, being totally mechanical, consists of a large number of moving parts, is maintenance intensive and is a batch process. Superior methods of chopping/decladding and dissolution and a high degree of automation in the operation of the entire head-end process including fuel charging and transfer and continuous dissolution are now being explored. These include



hydraulic transfer of the spent fuel, development of a single pin chopping facility, improved blade design for the integral fuel assembly and hull compaction.

## 4.2. Process systems

**a. Developing alternate methods for Pu reduction:** The partitioning of uranium and plutonium is currently being practised by employing uranyl nitrate. While the method is effective, it requires a large excess of uranyl solution, which loads the subsequent processing steps having to handle this uranium. Technique capable of in-situ reduction of process solution can overcome this drawback. The in-situ reduction technique, as well as the equipment specially designed and fabricated for this purpose, have been extensively tested. Some preliminary work has been carried out on this subject.

**b. Reduction of waste types and volumes:** The present methods and techniques employed in the reprocessing plants still generate a significant amount of waste of different levels and types. It is proposed to develop flow sheets, processes and techniques whereby the net production of this waste can be further reduced from the reprocessing plants operations. This will not only result in economy in waste treatment, but will also enhance the safety levels of the plant operations.

**c. Recovery of solvents:** The solvent undergoes degradation due to exposure to high radiation levels and harsh chemical environment. The solvent requires periodic washing with chemicals to remove the degradation products. This results not only in increased costs of operation, but it also generates high volumes of waste. It is therefore proposed to study and develop processes whereby these solvents can be purified by techniques like vacuum distillation and reused in the plant.

**d. Separation of actinides:** It is well known that separation of actinides from high level waste would reduce their long-term storage hazard in their immobilisation in the form of a vitrified matrix. It would also reduce the active surveillance period to a few hundred years. Laboratory scale studies for determining the suitability of various solvents and methods of actinide separation are being carried out. A plant scale facility, to test various flow sheets with actual high level waste from reprocessing plants, is now being planned.

**e. Flow sheet development for alternate fuels:** Current development in the field of reactor system design considers alternate fuel schemes utilising mixed oxides of uranium and plutonium, as well as fuel utilising a combination of thorium, uranium and plutonium. This aspect has led to the initiation of work on the development of the flow sheets and equipment for the dissolution and processing of these fuels.

**f. New solvent contactor:** A modified type of mixer settler contactor has been developed which utilises the static mixing principle by combining the air-lift and a mixing device. This contactor has performed well for the uranium processing facility and is virtually maintenance free due to the absence of any moving components for its functioning.

**g. Liquid transfer/metering devices:** To meet the requirements of higher capacity plants in the future, high capacity airlifts with a combined on-line flow rate measurement facility have been developed and extensively tested with encouraging results.

**h. Remote maintenance:** The current design philosophy follows the direct maintenance approach which leads to unproductive plant outage and generation of additional waste and man-rem exposure for repair jobs. Hence, attempts are made to develop the concept of an integrated remote maintenance which calls for a new approach to the in-cell layout of equipment. For example, critical components like pumps, valves etc. which require frequent in-service inspection and maintenance can be installed in hot cells equipped with viewing windows, servo manipulators and robotic system to facilitate maintenance remotely. Provision, to a limited extent, has been made in the new plant at Kalpakkam.

**i. Modelling and simulation:** Optimal design and efficient control requires better understanding of the process. This can be achieved through closer process modelling and simulation. It is proposed to explore various computerised modelling and simulation techniques to afford better agreement between theoretical process models and observed performance data. Further, since there is a continuous need for skilled and trained operating man power, it is proposed to build a *training*

*simulator* to provide hands-on training to fresh entrants without causing undesirable interference with the existing plant.

## 5. SPENT FUEL STORAGE

India at present has mainly three types of reactors in operation in the context of fuel storage, i.e. thermal research reactors, light water power reactors of BWR type and pressurised heavy water reactors (PHWRs). Current practice in India is to store the spent fuel under water at-reactor (AR) or away-from-reactor (AFR) for cooling prior to their transportation to and processing in the reprocessing plant. Extensive experience now exists in the design, construction, operation and maintenance of the spent fuel wet storage facilities. In the light of India's option to recycle the plutonium in the thermal and fast reactors, the spent fuel storage for extended periods of more than 10-15 years is not envisaged.

India has participated in an IAEA co-ordinated research programme (CRP) on 'Irradiation Enhanced Degradation of Materials in Spent Fuel Storage Facilities'. The study of the corrosion mechanism and behaviour of materials of the fuels and pool becomes important when the fuel has to be stored for extended periods.

The major conclusions as a result of the CRP are summarised below:

- Al-clad fuel is susceptible to crevice or pitting corrosion under conditions of galvanic coupling even in a benign water chemistry storage. In contrast, Zr-2 and stainless steel 304/304L are resistant under such storage conditions;
- Existing crevices (e.g. between Al-clad fuel and storage racks) between two storage racks can operate under even low conductivity water. This would be accelerated by galvanic contact (e.g. with stainless steel racks);
- Stress corrosion cracking (SCC) of Zr-2 and of stainless steel pool lining is not a probability during wet storage;
- The electro chemical potential (ECP) values show change in ionic concentration around stored spent fuel which shifts the ECP to more active values. So, the uniform corrosion rate is higher in presence of radiation, e.g. for aluminium/zircaloy claddings, compared to those in the absence of radiation;
- ECP values show wide variation with surface conditions of materials, water chemistry and also time period of immersion. A long-term study may be needed to ascertain this observation.

## 6. WASTE STORAGE

Capacity of up to five years of interim storage is provided for the high and intermediate level liquid waste arising from the reprocessing facilities. The acidic waste is concentrated and stored in stainless steel high integrity tanks in underground vaults. The intermediate storage capacity requirements for future facilities would diminish in view of the planned co-location of the immobilisation facility with the reprocessing facility. Indian experience with the waste storage tanks has been highly satisfactory. The storage vault internals and surrounding are regularly monitored to ensure the integrity of the containment. Periodic agitation of the contents prevents sedimentation and cooling provision for the high level waste tanks maintains temperatures below 40°C to keep corrosion levels low. Some of the salient features for the waste storage facilities are:

**a. Containment:** The following barriers are provided in the design of waste storage facilities:

# of Barrier	Barrier
First Barrier:	The SS tank shell
Second Barrier:	SS lining for vault
Third Barrier:	Reinforced concrete containment (RCC) vault
Fourth Barrier:	Peripheral RCC envelope for vault

The fourth barrier i.e. the peripheral envelope for the vault is provided to enable detection of seepage into or from the vault. This envelope has a sampling and pumping facility. An array of bore holes with sampling provisions are provided to monitor contamination of sub-soil water body as a result of the highly improbable event of breach of all the four barriers/containments. An infiltration ring and well with a pumping arrangement shall be provided.

**b. Off gas system:** All the waste storage tanks constituting the primary containment are maintained at a negative pressure. The process air introduced in the system for sparging, instrumentation as well as condensable and non-condensable vapours and gases are exhausted from the tanks by means of an off gas system comprising scrubber, condenser, heater, fiber glass filters and blowers. Stand-by off gas filters and blowers are provided to ensure the availability of the off gas system at all times.

**c. Process cooling:** Continuous supply of process cooling water is provided to take away decay heat from the waste and maintain low temperature to minimise corrosion. Two loops are provided in the cooling water system, i.e. a primary and a secondary loop. The primary process cooling water (PCW) system is a closed loop of de-mineralized water in which decay heat from waste is absorbed and then transferred to the secondary PCW system through heat exchangers. The heat is ultimately dissipated through cooling towers.

**d. Ventilation and exhaust system:** Concrete vaults housing the waste storage tanks are exhausted through high efficiency particulate (HEPA) filters banks.

**e. Instrumentation:** This system is required to continuously monitor tank levels, sump levels, density, temperature and pressure in the system. The radiation level and air activity of the storage tank are also monitored.

**f. Sampling system:** Provision for sampling of the tank contents is required for analyses of the contents. Samples from the tanks are drawn into sample bottles by means of a remotely operated sampling system.

## 7. CONCLUSION

The reprocessing technology in India aims at meeting the challenge to sustain a power generation programme based on plutonium or  $^{233}\text{U}$  fuelled breeder reactors. This calls for building reprocessing plants of larger throughputs during the coming decades. Simultaneously, the facilities for immobilising the waste arising from these plants are planned for the safe interim storage of the solidified waste before a repository site is chosen for the permanent safe containment. Towards this aim, it is essential to upgrade the reprocessing and waste management technologies to result in cost effective measures and reduced operator intervention. Some of these being the need to develop new materials for the critical in-process equipment like dissolver, evaporators etc. and full scale mock-up for performance evaluation of equipment prior to their introduction in the plant. Vitrification has been accepted as an appropriate technology for immobilising high level waste to meet the long-term safety requirements. This technology has now been successfully developed in India and it is now possible to carry out the vitrification operation concurrently, so that the waste product is also in the final solid form. This would lead to tremendous cost savings by co-locating the vitrification and reprocessing facilities. The technology has been undergoing an evolutionary change over the years and this has found expression in the improvements which reflect the state-of-the-art in each successive facility.



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## Abstract

In Japan, 52 nuclear power reactors are operating with a total power generation capacity of 45 GWe. The cumulative amount of spent fuel arising, as of March 1998, is about 14 700 tU. Spent fuel is reprocessed and recovered nuclear materials are to be recycled in LWRs and FBRs. Pu utilization in LWRs will commence in 1999. In January 1997, short-term policy measures were announced by the Atomic Energy Commission, which addressed promotion of the reprocessing programme in Rokkasho, plutonium utilization in LWRs, spent fuel management, back-end measures and FBR development. With regard to the spent fuel management, the policy measures included expansion of spent fuel storage capacity at reactor sites and a study on spent fuel storage away-from-reactor sites, considering the increasing amount of spent fuel arising. Valuable experience was been accumulated at the Tokai Reprocessing Plant (TRP), from the start of hot operation in 1977 up to now. The role of the TRP will be changed from an operation-oriented to a more R&D oriented facility, when PNC is reorganized into the new organization JNC. The Rokkasho reprocessing plant is under construction and is expected to commence operation in 2003. R&D of future recycling technologies is also continued for the establishment of a nuclear fuel cycle based on FBRs and LWRs.

## 1. INTRODUCTION

Japan has scarce energy resources and depends on foreign resources for most of its energy needs. Therefore, Japan has made efforts to utilize nuclear power since the mid 1950s, by carrying out research and development and to promote commercialization of the peaceful use of nuclear energy. Since its initial stage, the development and utilization programme has consistently called for the recycling of nuclear fuel.

Today, nuclear energy plays an important role as a key energy source and the nuclear power generation accounts for about 34 % of the total electric power generation.

This report reviews briefly Japan's basic long- and short-term policy on nuclear energy, and describes the current status and prospects of generation, storage and transportation of spent fuel. Some explanation is given on the research and development of spent fuel storage technology.

## 2. R&D ON NUCLEAR ENERGY, BASIC POLICY AND CURRENT STATUS & FUTURE PROSPECTS OF SPENT FUEL MANAGEMENT

### 2.1. Long-term programme for research, development and utilization of nuclear energy

The basic policy of nuclear energy is defined in the long-term programme (LTP) for research, development and utilization of nuclear energy [1]. The LTP is determined by the Atomic Energy Commission (AEC) of Japan and revised approximately every five years. According to the atomic energy basic law, it is required that research, development and utilization of nuclear energy are limited to peaceful purposes and that assurance of safety is the foremost consideration in them.

In 1994, the AEC revised the LTP. The programme intends to guarantee future energy security by steadily carrying forward research and development efforts aimed at future commercial commissioning of nuclear fuel facilities, involving the reprocessing of spent fuel and the recovery of plutonium and uranium to allow the reuse of these materials as nuclear fuel. Furthermore, recycling of nuclear fuel contributes to preservation of resources and environment, and to improved management of radioactive waste.

In the basic concept of LTP, the fast breeder reactor (FBR) is the kernel of nuclear power generation in the long-term, together with light water reactors (LWRs). Research and development is to be undertaken, in co-operation with the government and the private sector in order to establish a

commercial system of nuclear fuel recycling based on FBRs by around 2030. Also, construction of a commercial reprocessing plant and Pu utilization in LWRs are steadily promoted. Experience with nuclear fuel recycling in LWRs is considered to be important in order to establish a comprehensive technological system of plutonium utilization, which is indispensable to the above system based on FBRs.

Nuclear fuel recycling is promoted on the principle of not possessing plutonium beyond the amount required to implement the programme, i.e. the principle of no surplus plutonium, as well as having a very strict management of nuclear materials, coupled with transparency so as to provide assurances regarding adherence to non-proliferation of nuclear weapons.

## **2.2. Policies to promote the nuclear fuel cycle in the short term**

After the sodium leak accident in the secondary system of the FBR prototype "MONJU" in December 1995, the government made efforts to build a national consensus on the nuclear fuel cycle policy and to promote the disclosure of information and the participation of the general public in the policy decision making process.

The AEC has deliberated and decided short-term concrete measures of policy concerning the nuclear fuel cycle at the end of January 1997, taking into account the outcome of the discussion by the Advisory Committee for Energy which is an advisory body to the Minister of International Trade and Industry (MITI). The measures were consented by the Cabinet in February 1997 and are committed to steady promotion of the reprocessing programme for the plant under construction in Rokkasho, as well as to the promotion of the nuclear fuel cycle through the following policy measures in the short term:

- 1) Plutonium utilization in LWRs**
  - Start the utilization of MOX fuel in three or four reactors by 2000, expanding the use of MOX fuel to ten-odd reactors by around 2010;
- 2) Spent fuel management**
  - Store spent fuel appropriately as an energy source until reprocessed. Immediate measures are necessary in some existing nuclear power plants to expand their storage capacities with the understanding of local public;
  - Initiate a study aiming at an early conclusion on the development of the necessary environment to enable spent fuel storage at away-from-reactor sites by around 2010, in addition to the storage at reactor sites, given the increasing quantities of stored spent fuel in the long-term prospect;
- 3) Backend measures**
  - Present a total vision of disposal measures aiming towards the smooth implementation of final disposal of high level radioactive waste, through a broad range of discussions on the social and economic aspects;
  - Put in place the institutional infrastructure necessary for decommissioning nuclear facilities;
- 4) Development of FBRs**
  - Discuss future FBR development strategies, including treatment of MONJU, by a special committee on FBRs established under the AEC.

In March 1997, a fire and explosion occurred at the Bituminization Demonstration Facility of Tokai reprocessing plant of the Power Reactor and Nuclear Fuel Development Corporation (PNC). Investigation of this accident is being intensively carried out. PNC will be reorganized and formed into a new body "Japan Nuclear Cycle Development Institute (JNC)" from 1 October 1998.

In June 1997, the chairman of the AEC announced a statement which reconfirmed the cabinet consent of February 1997, i.e. the establishment of the nuclear fuel cycle should be promoted as before.

## 2.3. Current status and future prospects of spent fuel reprocessing

Reprocessing service will be provided by the Tokai reprocessing plant, the Rokkasho reprocessing plant and foreign reprocessors (contracts with BNFL and COGEMA). The Tokai reprocessing plant, which has an annual reprocessing capacity of around 90 t U, will shift its major role to research and development of future reprocessing technologies after the reorganization of PNC to JNC.

The Rokkasho reprocessing plant, Japan's first commercial reprocessing plant, will have an annual reprocessing capacity of 800 t U, and is scheduled to go into operation in 2003. The reprocessing capacity and technology of the second commercial reprocessing plant will be decided around 2010.

Spent fuel arisings exceeding the available reprocessing capacity will be properly stored and managed as an energy stockpile, until reprocessed.

## 3. SPENT FUEL ARISING, STORAGE AND TRANSPORTATION

### 3.1. Spent fuel arising and transportation

At the end of August 1998, 51 commercial nuclear power units and one ATR were in operation in Japan, and their total electric power generation capacity is about 45 GWe. According to the report by the Electric Utility Industry Council, a government advisory organization, nuclear power generation capacity will increase to 70 GWe in 2010. The cumulative amount of generated spent fuel was about 14 700 t U (about 13 200 t U from LWRs and 1500 t U from GCRs), as of March 1998.

The yearly production of spent fuel is currently about 900 tU/a and it is estimated, from the projected power generation capacity, that the annual generation rate of spent fuel will be 1400 and 1900 t U/a by the years 2010 and 2030, respectively (see Table I).

TABLE I. PROJECTED ANNUAL SPENT FUEL ARISING IN JAPAN  
(ton U)

Year	1998	2010	2030
Annual Spent Fuel Arisings	900	1400	1900

Part of the spent fuel generated in Japan has been transported to reprocessing plants. The shipment of LWR spent fuel to British and French reprocessing plants started in 1973 and the final shipment took place in 1998, resulting in the accumulated shipment of 5 610 tU. The shipment to the Tokai reprocessing was 940 t U up to now.

The initial shipment to the Rokkasho reprocessing plant started in October 1998 and the scheduled amount of spent fuel shipped to Rokkasho, until the reprocessing plant will be in operation in January 2003, will be 1600 t U.

### 3.2. Spent fuel management

Before reprocessing and recycling all spent fuel, spent fuel will be stored as an energy stockpile until the time it can be reprocessed. In view of the long-term increasing amount of stored spent fuel, MITI, STA and utilities studied the introduction of off-site storage of spent fuel and issued a report in March 1998, which pointed out the necessity of introducing such storage facilities by around 2010 [2]. In June 1998, MITI's Subcommittee on Nuclear Energy of the Energy Council published the report which recommended the construction of an interim storage facility for spent fuel as a future source of fuel (i.e. for recycling). Currently, the relevant government organizations are preparing the necessary institutional arrangements for introducing such facilities (see Table II).

TABLE II. REQUIRED AMOUNT OF INTERIM STORAGE FOR SPENT FUEL

tU

Item	F.Y. 1997~2010	F.Y. 2011~2020	F.Y. 021~2030
Produced spent fuel (a)	15 200	16 000	19 100
Dispatched to Rokkasho (b)	7 600	8 000	8 000
Dispatched overseas (c)	70	-	-
Storage at power plant site (d)	5 200	4 200	2 500
Required amount of interim storage (a-b-c-d)	2 300	3 800	8 600
Accumulated amount	2 300	6 000	14 600

Note: F.Y. stands for Fiscal Year

#### 4. REPROCESSING OF SPENT FUEL

##### 4.1. Tokai reprocessing plant

###### 4.1.1. History

The reprocessing project of PNC started in September 1956, when the Atomic Energy Commission of Japan decided that reprocessing of spent fuel and treatment of radioactive waste should mainly be done by the Atomic Fuel Corporation (AFC), the predecessor of PNC. In 1959, an Advisory Committee for reprocessing was formed within the AEC to formulate a guideline for development of the reprocessing technology. In conjunction with the recommendations put forward by a survey team which visited overseas reprocessing plants, a decision was made to construct a pilot reprocessing plant using the advanced technology by other countries.

In 1963, the AFC had contacted the Nuclear Chemical Plant (NCP) of UK for a preliminary design of the plant and in 1966, the Societe Generale pour les Techniques Nouvelles (SGN) of France started with a detailed design. Since 1968, in parallel with the ongoing detailed design, the governmental licensing procedure had been followed and permission for plant construction was granted by the Japanese Government in 1970.

Plant construction started in 1971 as a joint venture between SGN and JGC (Japan Gasoline Corporation). The plant was completed in 1974 and hot testing started in September 1977 after completing the tests with unirradiated uranium. The operational license was given after passing the Governmental inspections by the end of 1980.

###### 4.1.2. Amount of reprocessed fuel

From the start of hot operation on 22 of September 1977 until the end of March 1997, the total amount of reprocessed fuel from LWRs and the ATR Fugen (Advanced Thermal Reactor using heavy water as moderator) was 935.9 t U. Of this amount, 10.4 t MOX fuel from the ATR Fugen was reprocessed providing valuable experience with MOX fuel reprocessing. The maximum burnup of the spent fuel assemblies was 35 GW·d/t (see Tables III, IV).

The amount of plutonium nitrate recovered as a final product was about 6.4 t, and most of the Pu has already been sent to Pu conversion plant for use at the ATR Fugen, the experimental FBR Joyo, and the proto-type FBR Monju. Part of uranium trioxide powder, recovered as final product, has been sent to the Ningyo-touge site of PNC for conversion to UF<sub>6</sub>. From January 1995, part of the concentrated high level radioactive waste has been sent to the Tokai Vitrification Facility (TVF) for vitrification.

TABLE III. AMOUNT OF REPROCESSED FUEL AT TRP

Fuel Type	No. of FAs	t U
BWR	3 275	582.1
PWR	814	301.7
ATR*	339	52.0
Total	4 428	935.9

\* including 41.5 t UO<sub>2</sub> fuel

TABLE IV. YEARLY REPROCESSED AMOUNT OF SPENT FUEL AT TRP

F.Y.	77	78	79	80	81	82	83	84	85	86	87	88	89	90	91	92	93	94	95	96	97
	8.0	11.1	11.9	54.7	53.0	33.4	1.9	5.2	73.5	69.2	51.4	19.0	49.1	85.9	81.7	71.0	37.0	95.7	51.4	71.5	0.0

t U

*4.1.3. Major maintenance activities [3]***(1) Remote repair of dissolvers R10 and R11**

In April 1982, a small amount of radioactivity was found in the steam condensate from a dissolver. After confirming that one of the two dissolvers (R11) had small defects which consisted of pin holes in the welded part on the barrel of the dissolver, operation was resumed by using the other dissolver (R10) until February 1983, when the dissolver R10 showed the same kind of defects. The remote repair technology had been developed and from September to November 1983 the in-situ repair of the two dissolvers was successfully carried out for the first time in the world.

**(2) Installation of a new dissolver R12**

Leakage in the two dissolvers occurred rather unexpectedly and subsequently the third dissolver was installed in a spare dissolver cell. The new dissolver R12 was fabricated with improved material and welded lines were eliminated from the inside steam jackets. The fabrication of the dissolver was finished in April 1984 and the dissolver was installed by the end of November 1984.

**(3) Repair of acid recovery evaporator**

During the final stage of hot testing in August 1978, a minor leak was detected in the acid recovery evaporator which was caused by pin holes in the welded part of the heating tube. The whole evaporator was exchanged after decontamination and dismantling of the leaking evaporator by the end of December 1979. However, the new one leaked also in February 1983, caused by corrosion of the heating tube and at that occasion only the boiler part of the evaporator was replaced with domestic produced materials. The reparation period was seven months which was shorter compared to the former one.

*4.1.4. Scheduled shut-down of plant operation*

The operation of the TRP became steady and stable since 1985, after many modifications and improvements, however, the requirement of increasing the amount of spent fuel for reprocessing at the TRP is stronger than before, because of the demand for more plutonium for the ATR and FBR fuel cycle development. The design capacity of the TRP is 0.7 tons per day and the operation licence permits the TRP to reprocess up to 210 tons per year. However, it is difficult to reach this maximum, because of yearly regulatory inspections, the physical inventory taking (PIT) of nuclear material and the periodical maintenance.

The total operable days per year of the TRP have been calculated as about 170 days and, assuming an average plant efficiency of 60%, the derived yearly production of TRP would be about 70 tons.



In order to improve the production rate, the number of operable days should be increased on one hand and the plant efficiency improved on the other hand. The operable days per year were increased by shortening the maintenance and regulatory inspection periods. For improving the plant efficiency, it became clear that the fuel assembly shearing process and clarification process for dissolved fuel solution had to be modified. In the long range, it was also obvious, to prevent any plant outage caused by failure of the main equipment, that the acid recovery evaporator should be replaced and that modification of the fuel assembly shearing and clarification process should be made.

#### **(1) Replacement of the acid recovery evaporator**

The first acid recovery evaporator leaked after 6 000 hours of operation and leakage of the second one occurred after 13 000 hours. The material of the evaporators was 25% chromium and 20% nickel alloy of stainless steel and a conservative estimation was that the third evaporator would leak again after 13 000 hours of operation, which was expected around the middle of 1988. On the other hand, the development of corrosion resistant material was continuously carried as of the day the leak of the first evaporator occurred and it became evident that the 95% titanium and 5% tantalum alloy material showed a good corrosion resistance behavior in this corrosive environment.

The decision was taken to replace the third evaporator with a new one made of Ti-5%Ta alloy. This work was started in June 1988 and was performed smoothly within the scheduled 11 months period, based on the experience of the two earlier replacements.

#### **(2) Replacement of plutonium solution evaporator**

The design of the original plutonium evaporator was to connect the washing column to the boiler part with the flange and the material of the former one was stainless steel and of the latter one was corrosion resistant titanium. A pin hole defect appeared in the column part during 1982 after 10 000 hours of operation and in-situ repair was done. In year 1984, the replacement of the whole evaporator was done after 12 000 hours of operation.

A decision was made to replace this evaporator because of its 9 000 hours of operation and the material of the column part chosen was Ti-5%Ta alloy in order to prolong the operational life. The improvement was made to remove the flange connection by welding the titanium and the Ti-5%Ta alloy. The replacement was done in the cell within 3 months.

#### **(3) Modification of boiler part of acid recovery distillator**

The acid recovery distillator was fabricated from stainless steel and in February 1981, a corrosion leakage occurred in the part of the heating coil after 13 000 hours of operation and repair work was done within 1.5 months. In 1984, the boiler part of the distillator was replaced within 4 months. A new distillator was installed to replace the old one and has separable heating tubes in the boiler part of the distillator for easy maintenance.

#### **(4) Modification of fuel assembly shearing machine**

Many modifications for internal parts of the shearing machine were done to improve the operability and maintenance ability.

#### **(5) The addition of a second pulsed filter**

The clarification method of the TRP was done through a pulsed filter. The filtration of the dissolver solution clogs the sintered stainless filter gradually and finally it will necessitate the replacement of the filter cartridge affecting the plant. To improve the plant efficiency, a second pulsed filter was added in the clarification process. A new type of valve for changing the use of one filter to the other was developed and installed inside the cell for easy maintenance and high fidelity.

The modifications inside the cell were done after tedious decontamination of the equipment and piping and the permissible working time was limited because of the still rather high radiation dose. The total installation work took more than one year after a delay of 4 months for final modifications.

#### *4.1.5. Evaluation of major modifications on plant performance*

The scheduled shut-down of plant operation for the modifications lasted 15 months. Around 500 persons of PNC were involved in this work and the number of contracted workers of constructor and engineering firms were about 1 600 (100 000 man-day). The accumulated radiation dose was 5 man·Sv (500 man-rem).

The original intention of the plant improvement was to increase the yearly amount of processed fuel from 70 tonnes to 90 tonnes. At the end of September 1989, operation recommenced after the major modifications were done and after one year, in September 1990, the reprocessed amount was 83 tonnes of spent fuel and during the period from January to the end of November 1990, 99 tons of fuel were reprocessed.

#### *4.1.6. Radiation exposure control of plant personnel*

To minimize exposure and avoid excessive exposure of an individual in the plant, the investigation levels for exposure are set over three months, e.g. 3mSv for effective dose equivalent. Measurements of radiation fields are conducted for the purpose of avoiding excessive exposure of personnel and confirming that working environment is satisfactory of operations. Exposure rates and concentrations of airborne radioactive materials are measured continuously by the automated monitoring system, signals of detectors are centralized into the health physics panels in the safety control room. The annual collective dose equivalent was around 1 man·Sv during normal TRP operation.

#### *4.1.7. Activity discharge from the plant*

Under normal operation of the TRP, low level radioactive effluents are discharged to the atmosphere and the ocean under rigid control. The radiation exposure to the public around the plant has been estimated for the potential pathways with the site specific parameters such as food consumption, concentration factors of marine organism and meteorological condition. External exposure, due to gamma ray from Kr-85, and internal exposure, via inhalation and oral intake of radionuclides, are evaluated for the airborne effluent. External exposures to contaminated fishing net and fishing boat are considered as pathways for fisherman. External exposure to contaminated beach and internal exposure via oral intake of marine products are evaluated for the liquid effluent.

The estimated annual effective dose equivalents are less than 0.1 percent of the annual effective dose equivalents limit for the public recommended by the International Commission on Radiological Protection (ICRP) since the operation of the TRP was started in 1977.

The amount of  $\beta$  discharged to the sea was reduced by additional installation of liquid treatment evaporators and atmospheric I-129 discharge is adequately controlled by installation of silver impregnated filters for trapping iodine.

The results of environmental monitoring, including the values of effluent discharge, are submitted to "Central Evaluation Advisory Committee for Environmental Radiation Monitoring", which is the advisory group of the Nuclear Safety Commission (NSC), for assessment of the monitoring results. The assessment results are reported to the NSC and are published in NSC's periodical. In 1991, the monitoring of C-14 atmospheric discharge was included in the monitoring programme of the TRP, because of its not negligible effects compared with other nuclides, after the licensing procedure of installation of a new marine discharge pipe line and evaluation of the C-14 discharge.

#### *4.1.8. Recent major incident at the TRP*

##### **4.1.8.1. Failure of highly active waste evaporator**

On 19 October 1995, a small amount of radioactivity was found in the steam condensate from the highly active waste evaporator. After confirming that one of heating tubes had a small defect, operation was resumed using the stand-by evaporator.

##### **4.1.8.2. The fire and explosion incident of the bituminization facility [4]**

A fire and explosion incident occurred at the bituminization demonstration facility (BDF) of the Tokai reprocessing plant on 11 March 1997. The BDF, which was designed in 1976-1977, started with hot tests in 1982. The operation of the BDF was stable and produced about 30 000 drums of products until the incident.

###### **(1)The process of BDF**

The waste in the receiving tank is transferred to the reactor vessel for every batch. Chemicals are added to the reactor vessel as a pretreatment of the waste. Sodium carbonate and barium hydroxide are added for the coprecipitation of strontium. Nickel sulfate and potassium ferrocyanoid are added to the reactor for the coprecipitation of cesium. To decrease the volatilization of iodine, sodium iodate, sodium sulfite and silver nitrate are added to the reactor. The waste is transferred to the feeding vessel and fed into the extruder via a double air lift. The feed rate of liquid waste is normally 200 l/hr.

In the extruder where fresh bitumen is fed, the waste is dehydrated and the mixture of waste and bitumen is kneaded. The resulting mixture is poured into a drum on the turntable. One drum is produced in about two hours. About ten drums of bituminized waste are produced in a batch.

###### **(2) Incident Sequence**

The sequence of the incident was as follows:

- 10:06 An operator noticed a flash in the filling room and observed a 2 m high pillar of fire from a drum. After several tens of second, all the drums seen through a shielding window emitted the pillar of fire. Hence, the fire and temperature detectors gave alarm.
- 10:08 The fire and temperature detectors gave alarm again.
- 10:10 An auto fire alarm was given. The operator did not see the fire because the inside of the cell was filled with smoke.
- 10:12 Water spray started to extinguish the fire. After 1 minute, the operator stopped the water spray since he did not see the fire in the cell. He saw the smoke leaked from the penetration of the manipulator.
- 10:13 The beta dust monitor of the conveyer room gave alarm. In the control room, ventilation alarm was given. An operator pushed the emergency stop button of the extruder.
- 10:14 (to 10:18) The beta dust monitors in the maintenance operation and cask storage room gave alarm.
- 10:18 The indicator of the cell ventilation blower in the control room showed that the air flow had stopped and that only an air-supply blower and an exhaust blower for the building ventilation operated.
- 10:23 All the blowers were put off.
- 20:04 An explosion occurred. The gamma area monitor in the maintenance operation room gave alarm. After the explosion, the thermocouples for the pouring tube indicated a doubling in temperature during two and half hour.
- 23:30 According to the monitoring camera outside the building, smoke stopped flowing from the window.

### **(3) Environmental effect**

The radioactivity released from the BDF was studied based on the two methods: one being the migration behavior from the fired drum through the building, the other the environmental monitoring. The released activity to the environment was estimated to be in the order of  $10^{-3}$  GBq for alpha nuclides and several GBq for beta nuclides. The committed dose equivalent was  $10^{-3}$  to  $10^{-2}$  mSv of magnitude.

### **(4) Cause of incident**

The chemical analysis of liquid waste and the bitumen mixture made during the course of the investigation so far,, did not show that the material was the cause of the fire. The findings related to the cause of the fire were the identification of fire in the 29th and 30th batch, functional group due to the oxidation reaction of bitumen observed in the 29th batch product and pouring the bitumen mixture into a drum with temperature higher than at normal operation. The experimental result, that the bitumen mixture caught fire after emission of the mist of high-boiling-point hydrocarbon, was useful for the explosion study.

According to the study on the cause of the fire and the explosion, possible causes of fire are the followings: (i) temperature rise in the bitumen mixture due to friction heat generated in the extruder, (ii) temperature rise in a filled drum by decrease in thermal conductivity and by exothermic oxidation reaction due to involving air in the extruder operated at a feed rate 20% lower than the normal condition, (iii) acceleration of oxidation/reduction due to the catalytic effect of the sludge. As for the cause of the explosion, it is estimated that several tens of species of high-boiling-point-hydrocarbon mist were generated from the drum on the turntable, the filling room was filled with the mist and exploded.

### **(5) Lessons learned from the incident**

There was a problem in operation associated with the fire extinguishing and the ventilation system. It took 6 minutes after detecting the fire to extinguish the fire by an operator with the water spray, subject to an order of the manager. When spraying water, the air supply to the cell ventilation system was not stopped against the operation manual of the water spray. The spray was stopped after a minute because no fire was observed through the shielding window. This operation lead to the insufficient fire fighting.

No countermeasure was considered in the design of ventilation system in case of plugging of the filter and air supply to the ventilation system. The ventilation system was stopped for a long time. These were possibly attributed to the explosion and personnel dose.

### **(6) Items to be modified in the facility**

To prevent fire, the following is required:

- since the differential thermal analysis can not measure the heat generation of the waste sample, a differential scanning calorimeter is necessary;
- the temperature monitoring system of the extruder should be modified to control and measure the temperature of the bitumen mixture;
- the monitoring system for the drum after filling the bitumen mixture should be modified.

To detect and extinguish fire, the following equipment is necessary:

- a recorder for the fire detection system;
- a display and continuous monitoring system for the temperature in the cell;
- an auto fire extinguishing system;
- an auto shut-off of air supply to the ventilation system when the water spray operates;
- a sufficient number of series of filter systems.

### **(7) Management of operation and education/training**

To avoid the problems identified in the study of the incident, the following should be improved from the viewpoint of management:

- modification of assignment of operation as a contract;
- attaching importance to operational division;
- clarifying the judgment and action in an abnormal event;
- thorough and prudent review of the operation plan;
- a well prepared education system, multi-event training and checking system.

#### **4.1.9. Future of TRP**

PNC will be reorganized and formed into a new body "Japan Nuclear Fuel Cycle Development Institute (JNC)" from 1 October 1998. The role of the TRP will be changed from an operation oriented to a more R&D oriented facility on this occasion. The main role of the TRP was to provide reprocessing services to utilities, but making the establishment of JNC as a turning point, the TRP will be used in the future as a hot test field for the R&D of MOX fuel reprocessing and FBR fuel reprocessing technology. Initially, the reprocessing of small amounts of high burnup fuel will be done at the TRP to obtain data, with supplementary data from a small laboratory (Operational Test Laboratory) at TRP. The improvement of the maintenance technique will also be done. For these R&D activities, the research plan will be reviewed externally and the R&D result will be disclosed based on information disclosure criteria with due regards to non-proliferation concerns.

### **4.2. Rokkasho reprocessing plant**

Japan Nuclear Fuel Ltd. (JNFL) has started the construction of a reprocessing plant with a capacity of 800 t U/year in April 1993, in Rokkasho Village, Aomori-Prefecture. The principal facility specifications are shown in Table V. JNFL applied for the authorization of reprocessing business through the STA to the Prime Minister in March 1989. The first step in the review by STA for reprocessing was completed in August 1991, and the second step by AEC and NSC was finished in December 1996.

TABLE V. PRINCIPLE SPECIFICATIONS OF THE ROKKASHO FACILITY

<b>Reprocessing plant</b>	Method	PUREX method
	Capacity	800 t U/year (average) 4.8 t U/day (maximum)
<b>Spent fuel storage pool</b>	Capacity	3 000 t U
	SF residual enrichment	less than 3.5 wt.%
	SF cooling time	over 1 year before receipt
	SF burnup	45 000 MW·d/t U (average) 55 000 MW·d/t U (maximum)

JNFL intended to modify mainly the purification process in order to rationalize the plant design. The modification required to do both review steps again. The first step was finished in December 1996 and the second step was completed in July 1997. The reprocessing plant will go into operation in January 2003.

### **4.3. FBR fuel reprocessing**

For the FBR fuel reprocessing, PNC had developed its own process and equipment as well as remote handling technology through large scale cold mock-up tests and laboratory scale hot tests. The chemical processing facility (CPF) has been used for hot process tests since 1982. The recycle equipment test facility (RETF) is for hot engineering-scale equipment test and is under construction. Based on the recent change in the situation, the plan for the RETF will be reconsidered.

## 5. RECYCLING OF PLUTONIUM AND URANIUM RECOVERED FROM SPENT FUEL

### 5.1. Future nuclear fuel recycling programmes

For a certain period, LWRs will continue to be a major source in Japan's nuclear power generation programme and some of them will recycle plutonium. FBRs will play a central role in the nuclear fuel recycling system in the future and will be the principal reactors to recycle plutonium in combination with LWRs.

The Rokkasho reprocessing plant will be the main source to supply plutonium to the LWRs and FBRs in the future. The reprocessing capacity and technology of the second commercial reprocessing plant will be decided in due course. Recovered plutonium in overseas reprocessing plants will be used to fabricate MOX fuel by overseas fuel manufacturers and then utilized in Japanese LWRs. It is necessary to construct a domestic commercial MOX fabrication plant for LWRs, taking into account the operation plan of the Rokkasho reprocessing plant.

### 5.2. Project of fast breeder reactor

The construction of MONJU, a loop type liquid metal FBR of 280 MWe output, was completed late in April 1991. The pre-operational test started in May of the same year. The test consisted of a function test and a start-up test. The function test was finished in 1992. The reactor had reached initial criticality in April 1994 and started generation of electricity in August 1995.

With regard to the sodium leak accident that occurred in December 1995, STA announced a final report in February 1997 and PNC completed the investigations of the cause of the accident in March 1997. A total safety evaluation of the MONJU plant is now being performed in order to improve its safety.

As the cabinet consent in February 1997 showed, further strategies for development of fast breeder reactor were discussed in the Special Committee on FBRs established under the AEC. The committee recommended the promotion of development of fast breeder reactors as one of the promising non-fossil energy sources for the future under a flexible programme while pursuing safety and economy.

### 5.3. Programme for utilization of MOX fuel in LWRs

Utilization of MOX fuel in LWRs is important from the view point of recycling plutonium before commercialization of FBRs. Corresponding to the Cabinet Consent, the Federation of Electric Power Companies of Japan announced a programme for the MOX utilization in LWRs in February 1997.

### 5.4. MOX fuel fabrication

Development of MOX fuel fabrication by PNC started at the plutonium fuel development facility (PFDF) in 1965. To fabricate fuel for FBRs and ATR, PNC has been operating the plutonium fuel fabrication facility (PFFF) and the plutonium fuel production facility (PFPF). A domestic LWR MOX fuel fabrication plant for commercial operation will have a capacity of around 100 t/y.

### 5.5. Utilization of recovered uranium

Recovered uranium can be converted to uranium hexa-fluoride followed by re-enrichment and re-conversion and can be mixed with enriched uranium or with plutonium to be recycled as MOX fuel. Re-enrichment is considered to be the best method of recycling uranium in terms of economy and the amount of usable uranium recovered. About 240 t of recovered uranium will be converted to uranium hexa-fluoride by PNC under a contract with Japanese utilities.

## 5.6. R&D for advanced nuclear recycling technology

For any future nuclear recycling system, it is not only important to strive for improvement of safety, reliability and economy, but also to pursue the possibilities of reducing the environmental impact and assuring nuclear non-proliferation. Long-term research and development will be conducted on advanced nuclear fuel recycling technology based on the FBR, such as recycling of new types of fuel and recycling plutonium together with actinides. R&D programmes on the advanced nuclear fuel recycling technology are being discussed in the AEC's Advisory Committee on Nuclear Fuel Recycling Programme.

## 6. CONCLUSIONS

Japan intends to guarantee future energy security, by steadily carrying forward research and development efforts aimed at future commercial commissioning of nuclear fuel recycling, involving the reprocessing of spent fuel and the recovery of Pu and U to allow the reuse of these materials as nuclear fuel in LWRs and FBRs.

Concerning spent fuel management, the policy measures included the expansion of storage capacity at reactor sites and to study the option of storage in facilities at away-from-reactor sites in addition to storage at-reactor sites.

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#### Abstract

Reprocessing of WWER-440 spent fuel is carried out at the RT-1 plant of the combine "PO MAYAK". In taking the plant into operation, the closed fuel cycle concept was realized in Russia. The plant capacity is 400 t/a which provides for complete reprocessing of the spent fuel (SF) from national and foreign WWERs-440. The paper provides information on WWER-440 spent fuel received by "PO MAYAK" from foreign and Russian NPPs. The unique equipment was designed and manufactured for the initial reprocessing operation, which is famous for high reliability and long service life. The final product of the plutonium stream is plutonium dioxide, which is stored, and that of the uranium stream is uranyl nitrate hexahydrate melt with 2-2.4% uranium-235 enrichment. Such uranium enrichment is achieved at the evaporation stage due to admixing highly enriched uranium obtained from reprocessing SF from transport, BN-350 and BN-600 reactors (Russian type of fast reactors). The melt is then used for RBMK fuel fabrication. WWER-1000 fuel assemblies cannot be reprocessed at the RT-1 plant, because of their significant distinction in mass, dimensions and fissile content compared to WWER-440 fuel assemblies. Therefore, a decision was taken in 1976 to construct the RT-2 plant for WWER-1000 spent fuel reprocessing. Before the RT-2 plant will be put into operation, the storage facility, which was commissioned in 1985, will function as a regional storage facility and is used for spent fuel accumulation. It is assumed that the storage facility will be filled by 2015, when the spent fuel radiochemical reprocessing plant is to be put into operation. The paper deals with the technologies of spent fuel reception, storage and preparation for chemical reprocessing at "PO MAYAK". The long-term operational experience shows that the requirements for protection and nuclear and radiation safety were properly accounted for at the design, construction and operation stages of the RT-1 plant. The paper considers also the prospects of plutonium management.

#### 1. INTRODUCTION

WWER-440 spent fuel reprocessing is carried out at the RT-1 plant of the combine "PO MAYAK". The RT-1 plant has a 400 t/a capacity which allows to reprocess the WWER-440 spent fuel (SF) both at present and in the future. With the commissioning of this plant in 1976, the implementation of the closed fuel cycle was realized in Russia. Up to now about 3 000 t of spent fuel were reprocessed. Table I presents the data on the amounts of spent nuclear fuel from WWER-440 reactors which were transported from NPPs of other countries and Russia to "PO MAYAK" in the period 1979 to 1996.

TABLE I. AMOUNT OF WWER-440 SF TRANSPORTED TO "PO MAYAK"  
FROM OTHER COUNTRIES AND RUSSIA IN 1979 - 1996.

	Country	NPP	Spent fuel
1.	Armenia	Armenia	200.0
2.	Bulgaria	Kozloduy	295.0
3.	Czechoslovakia	Bohunice and Dukovany	98.0
4.	Finland	Loviisa	282.0
5.	Germany	Greifswald	178.0
6.	Hungary	Paks	168.0
7.	Russia	Kola	565.0
8.	Russia	Novovoronezh	412.0
9.	Ukraine	Rovno	244.0



The main product of RT-1 is enriched uranium. Besides, 2.6 t/a of civil plutonium is extracted at the plant. In 1996, the plutonium production decreased to 0.6 t. The total mass of the extracted plutonium stored at the "PO MAYAK" is about 30 t. Plutonium is reliably stored as dioxide and is intended for future use in fast reactors.

## 2. WWER-440 SF REPROCESSING

The technology for WWER-440 SF reprocessing is based on an extraction process known in the world as the Purex process. The process consists of SF dissolution with subsequent separation of plutonium and uranium from residual fission products and actinides for further recycle (see Fig. 1).

The spent fuel arriving for reprocessing is unloaded from container wagons into an interim storage facility. Then, the fuel is remotely transferred from the storage facility to the preparation cell for shearing. In the preparation cell, end fittings of fuel assemblies (heads and end pieces not containing fuel) are cut off using a special mechanism. The cut off parts are loaded into transport tanks and dispatched by a special vehicle on a linear asynchronous engine along the protected platform for long-term storage. The prepared fuel assembly comes to the shearing machine where it is flattened and cut into pieces.

The shearing machine is unique and has over ten units and components, the main one being the knife unit (see Fig. 2). The main operation is performed within the belly of the knife unit, that is, where nuclear fuel is opened and the structural materials of fuel assemblies are cut into small pieces of the given size. The Russian shearing machines use unique knife units. The service life of the knives is 230 thousand cuts. It is provided by a number of components the knife unit consists of and the use of a special shearing diagram. Because the shearing produces 1 - 2 % pieces with larger sizes than specified, the pieces falling down the slide are received by an inertial sieve of pre-set mesh sizes. The pieces remained on the sieve are placed by a manipulator into the knife unit for repeated cutting. The pieces of fuel rods passed through the sieve are poured into a batch operated dissolver where they are dissolved. All operations with fuel assemblies and end pieces are carried out remotely using different types of manipulators.

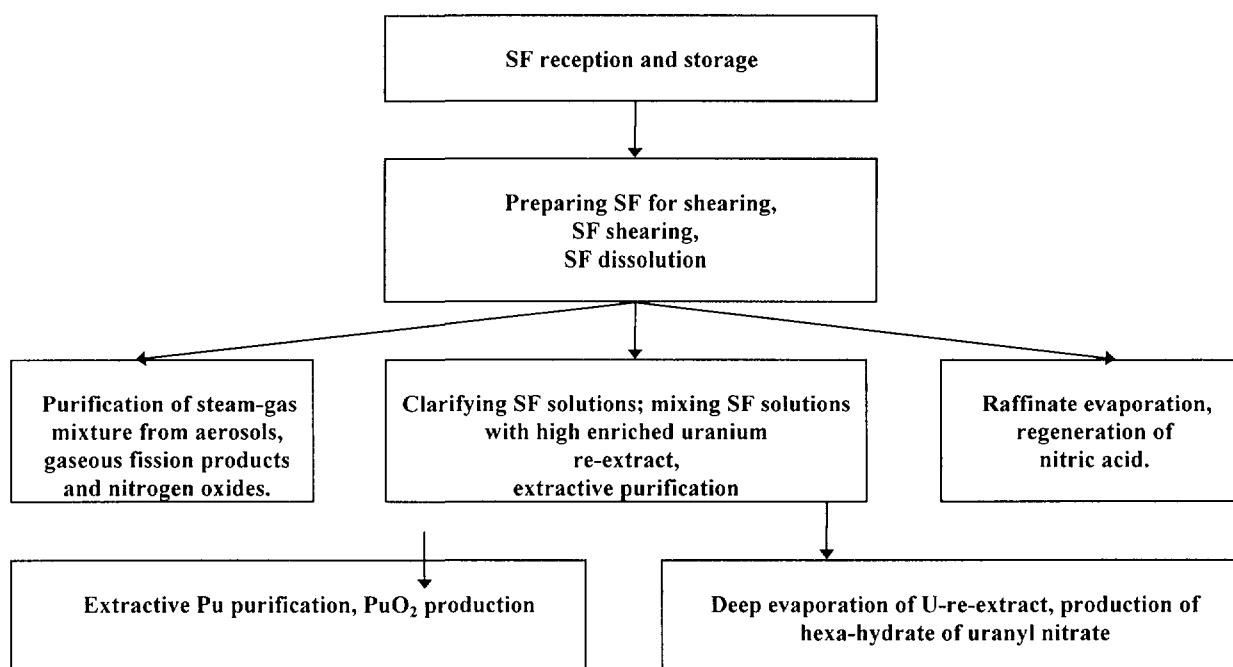
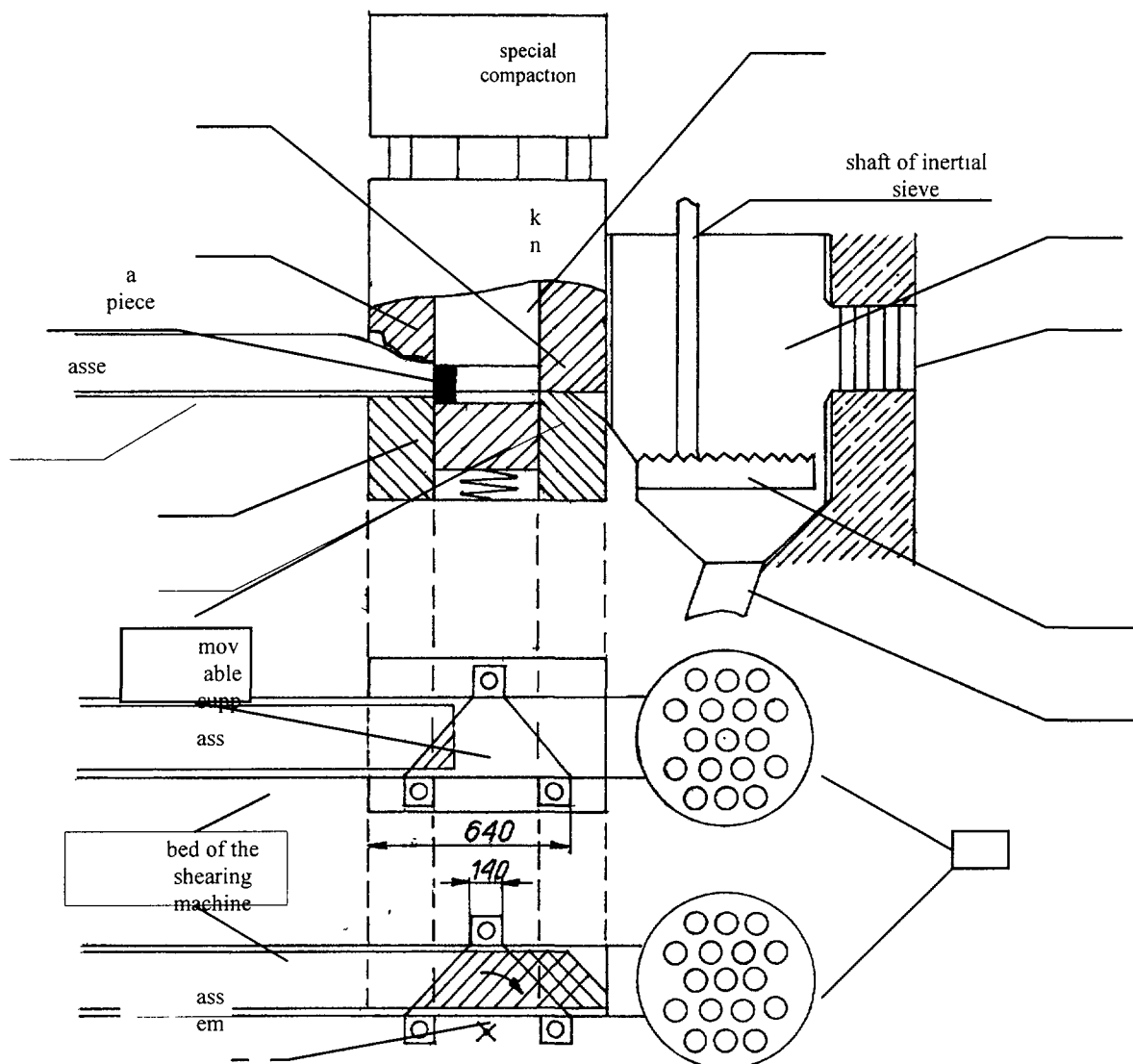


FIG. 1. PUREX process



a) Scheme of the knife unit, b) Section of flattened assembly clamped between the upper knife and movable support, c) Next section of a cut off piece of a flattened assembly according to scheme (b)

1 First clamp along the movement of the feed 2 Second clamp along the movement of the feed, 3 Upper knife, 4 Piece sorting chamber, 5 Cavitation nozzle window 6 Inertial sieve, 7 Chute 8 Possible axis of piece rotation 9 Bottom knife, another one along the movement of feed, 10 Bottom knife, the first along the movement of feed

FIG. 2. Scheme of the shearing machine

Pieces of claddings and other construction components are washed and, after control dissolution, withdrawn from the dissolver. By using a pulse pneumotransport, they are transferred through a pipeline to the storage bays for long-term storage where for fire safety reasons they are stored as a mixture with aluminium.

The SF solution is transferred for clarification and extractive purification from fission products. Uranium and plutonium purification from fission products and their separation are done in the 1<sup>st</sup> extraction cycle. Further purification is carried out in the separate uranium and plutonium streams. The total purification factors from fission products are:  $1 \times 10^7$  for uranium and  $1 \times 10^8$  for plutonium. The exposure dose rate from the end product is  $1.7 \times 10^3 \mu\text{R/s}\cdot\text{kg}$  for uranium and  $0.1 \mu\text{R/s}\cdot\text{kg}$  for plutonium. The end products from WWER-440 SF reprocessing are:

- Melt of hexahydrate of uranyl nitrate with 2.0 - 2.4 % enriched U-235 (due to admixing with highly enriched uranium received from reprocessing SF from transport and BN reactors). The melt is further used for fabricating RBMK fuel;
- Plutonium dioxide, to be stored for an interim period into a special storage facility. In the future the end product of the plutonium line will be MOX fuel for fast reactors.

The process at the RT-1 plant is 100 % remote controlled and monitored. The operations requiring scrupulous attention to their sequence and duration (valves, drives, etc.) have programmable control. The plant uses a three-zone layout of rooms with independent ventilation of each zone. The air from each vault goes through a tubular corridor towards the filter plant (a two step air purification) and is then vented to the atmosphere through a tall stack ( $H = 150$  m).

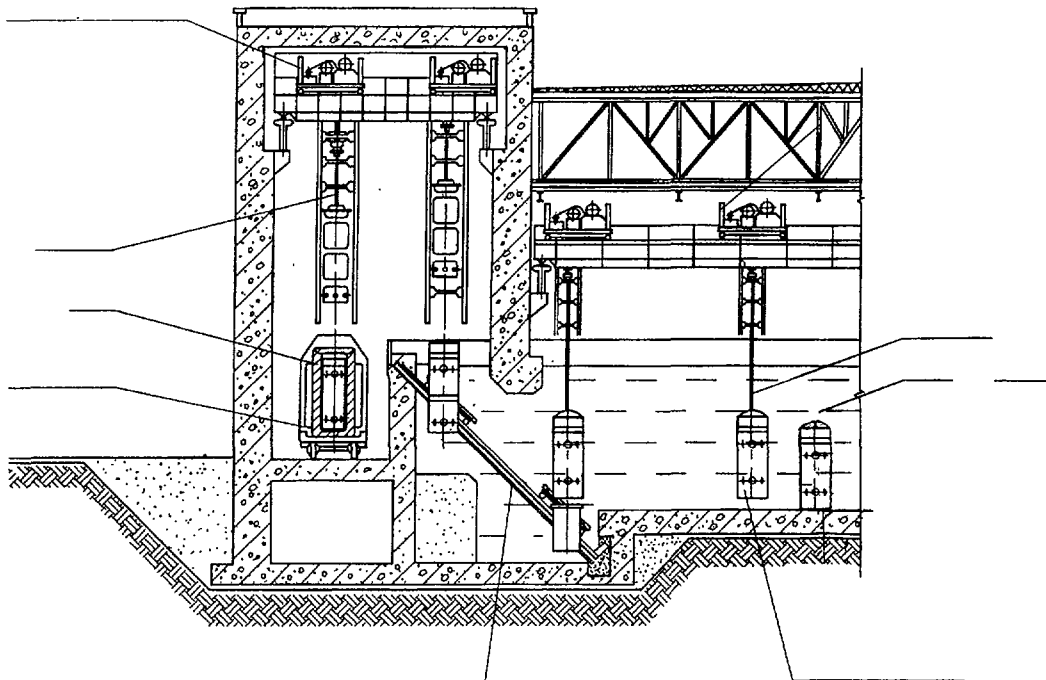
The air activity releases at the RT plant are as follows:

- $\alpha$  - release: 0.7 of the prescribed limit;
- $\beta$  - release: 0.2 of the prescribed limit;
- iodine - release: 0.07 of the prescribed limit.

The first zone accommodates the equipment with radioactive materials behind the biological shielding, the second zone the equipment for repair works and the third zone the operator and control rooms. In the rooms of different zones, a pressure drop of 5 - 10 mwg is maintained, so that at opening the access ports and doors between the zones, or at loosing the zone tightness, the air flow is directed from more clean to more dirty rooms. This allows the continuous control by not less than three barriers against radionuclide release into the environment. The equipment of short service life has redundancy. Failed pumps, control and measuring instruments, valves, etc. are remotely changed using special mechanisms without stopping the process.

### 3. FUEL HANDLING OPERATIONS

All the operations of the spent fuel reception and storage at the RT-site are remotely controlled. The sequence of fuel unloading operations is shown in Fig. 3. The railway car arrives to the transport corridor of the unloading area. The doors on the car roof are opened, the TK-6 transportation cask is depressurized (through a special ventilation system), the cask lid is unbolted and lid bolts are removed.



1 Crane; 2 Adapter; 3. Storage water pool; 4. Canister basket; 5. Elevator; 6. Railway car; 7 Cask; 8. Adapter; 9. Crane

*FIG. 3. SF reception and storage*

Further unloading operations are remotely controlled from the control board with the cask staying in the railway car. The observation of the unloading process is performed with TV cameras. A 15 t crane removes the cask lid, draws the filled baskets from the cask and puts them on an elevator of 15 t capacity which transfers the baskets to a storage facility. Here, a 15 t crane removes the basket from the elevator and places it in the storage position. Baskets with spent fuel are stored in water pools which have a capacity not less than the annual RT plant throughput.

#### 4. RADIOACTIVE WASTE HANDLING

This technological process of treatment of liquid radwaste (see Fig. 4) involves:

- radwaste collection, sorting, accumulation and intermediate storage;
- concentration;
- concentrates solidification;
- intermediate storage of solidified radwaste in special storage facilities;
- purification of gaseous and aerosol emissions.

The liquid high-level waste (HLW) resulting from spent fuel breakdown in the extraction cycles is transported for storage in stainless steel tanks, available at the plant site. Several groups of storage tanks with a capacity of 300 m<sup>3</sup> and 500 m<sup>3</sup> have been in operation for a long time.

The major engineering barriers necessary for safe operation of the tanks are the following:

- location of the tanks in stainless steel-lined vaults;
- equipping the tanks with a water-cooling system, which maintains the solution stored at a temperature level  $\leq 50^{\circ}\text{C}$ ;
- subsequent reduction of the hydrogen content to permissible standards by use of gas purification systems;
- issuing of solutions for treatment via special links with all necessary control means.

The collection and storage of HLW is a preliminary stage of waste conditioning. Later on, HLW is transferred to an accumulator tank for homogenization to the specified chemical composition and specific activity and then is vacuumed off to monte-jus. In these, the solution is adjusted to the specified chemical composition and mixed with fluxing additives. The flushed solution is dozed to an evaporator for twofold reduction of the solution volume. The evaporator bottom is periodically drained to a monte-jus and run off for reprocessing along with middle active solutions. The non-condensed gaseous phase goes to the gas purification system. The bottom from the accumulator monte-jus is continuously dozed to a direct arc furnace EP-500. Molasses (150 g/l of solution) is fed to obtain a reducing medium of nitrogen and ruthenium oxides and to decrease solid phase sweeping away to the gas purification system. In the furnace, the fluxed bottom solution goes through the stages of salt dehydration, oxide calcination and glass mass melting. Vitrification with phosphate glass production occurs at 1100 - 1150°C. The glass mass capacity of the furnace is 1.8 m<sup>3</sup>/d or 165 m<sup>3</sup>/a. The total activity of the glass mass thus obtained amounts to  $2.8 \times 10^8$  Ci/a.

The steam-gas phase, heated to 600 - 800°C, goes from the furnace to a bubbler-cooler for condensing and cooling. The condensate is collected and periodically fed for mixing with the initial solutions. The non-condensed gas phase goes to the gas purification system. Filled flasks are transferred from the glass mass dispensing cell (equipped with local suction) to the cell of canister completing. Here, the flasks are remotely loaded into the canisters and the canisters are welded. Then, the canister is drawn from the cell into the shielded container and transported to the storage facility located in the same building. The facility consists of ferro-concrete blocks with tubular wells to accommodate canisters with vitrified HLW (see Fig. 5). The construction of such facility is designed for receiving HLW with an initial heat release of 5 kW/m<sup>3</sup>. Loading/unloading operations with canisters are done using a three-item shielded container, which is transferred by a 120 t crane in the assembly hall. The facility design provides both for forced and natural cooling of filled canisters. The storage period may range up to several decades when needed.

The equipment for HLW vitrification and storage is located in two buildings linked with each other by a technological platform. The processes are 100 % automated and remotely controlled from the control panel.

In conclusion, the technology allows to obtain end products which meet the requirements of the National Standards and provide for environmental and personnel safety in compliance with the ALARA principle.

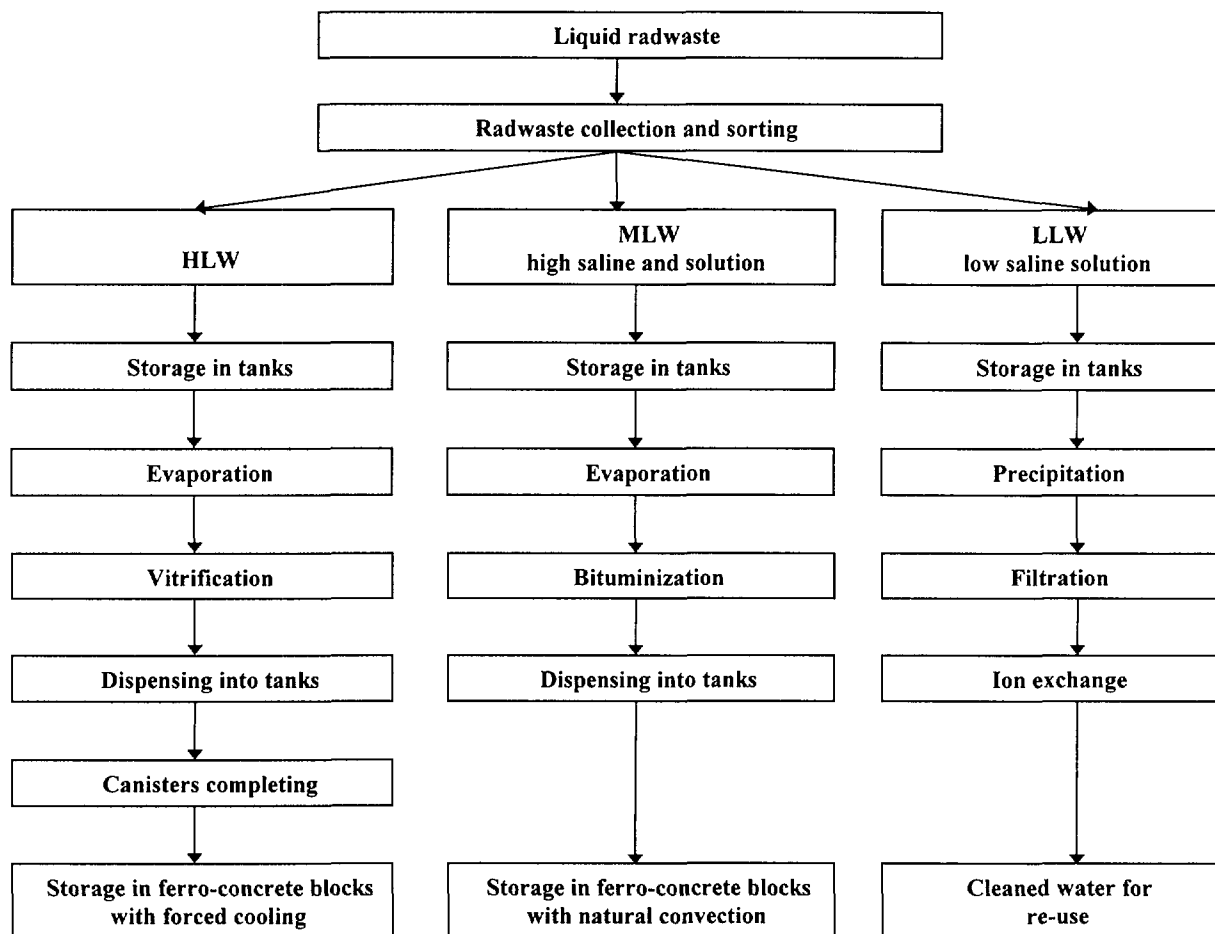
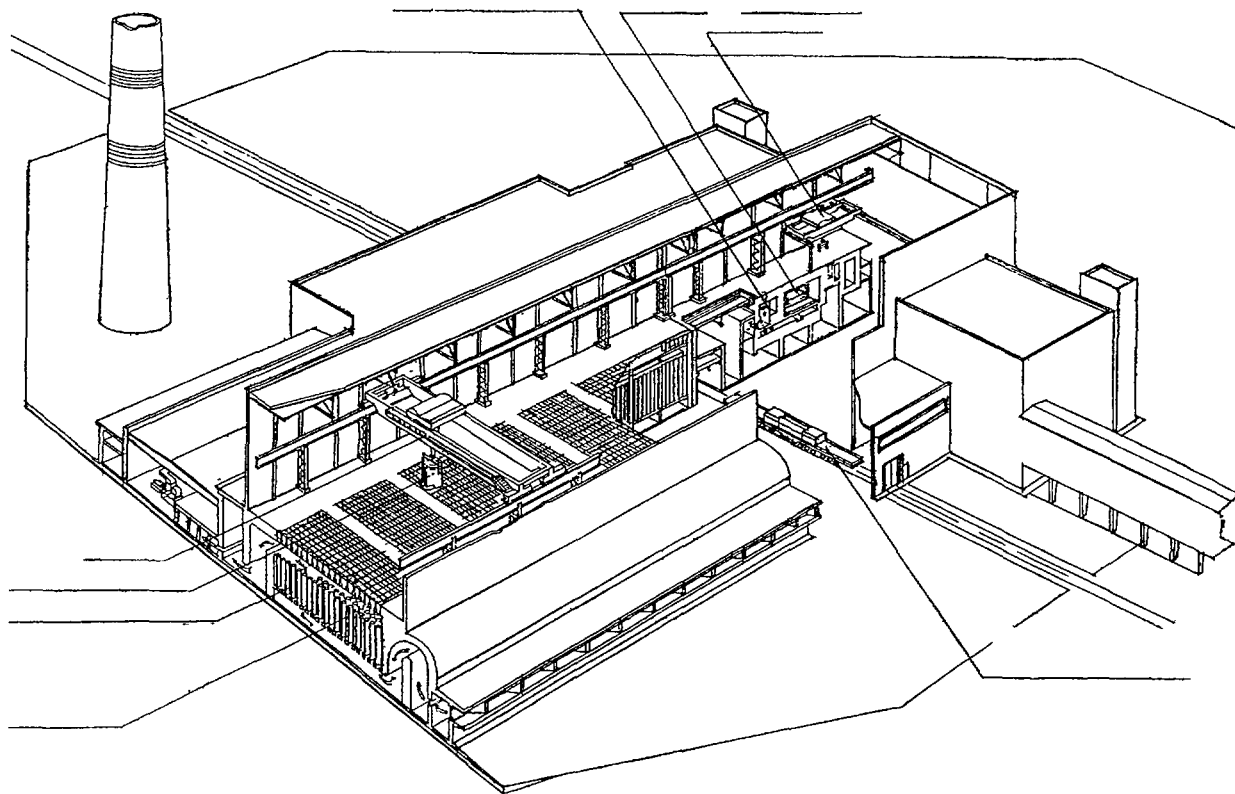


FIG. 4. Treatment of liquid waste

## 5. WWER-1000 SF REPROCESSING

WWER-1000 fuel assemblies cannot be reprocessed at the RT-1 plant because of significant differences in mass, dimensional characteristics and fissile material content compared to WWER-440 fuel assemblies; therefore, the decision was taken in 1976 to construct the RT-2 plant for WWER-1000 SF reprocessing. The reprocessing plant capacity will be 1500 t fuel per year and the plant is supposed to be commissioned by 2015. The RT-2 plant will use the well proven Purex process, though some modifications and upgrading will be introduced.

Before commissioning the RT-2 plant, a storage facility has been constructed and put in operation in 1985. The design capacity of the storage facility is 6 000 t U and it serves as a regional storage facility to accumulate SF. As of 01.01.99, 2 240 t U was stored in the storage facility and by 2005 the storage facility is supposed to be filled. There are plans for expanding the storage capacity in due course. Table II shows the accumulation dynamics for SF arrived from Russian and Ukrainian NPPs to the RT-2 storage facility.



1. Shielded cell; 2. Electric furnace; 3. Crane; 4. Railway car; 5. Canister; 6. Supports for canisters; 7. Shielded container; 8. Crane

FIG. 5. Storage facility for vitrified HLW

TABLE II. SPENT FUEL ARISING IN RT-2; SPENT FUEL SHIPMENTS FROM WWERS-1000

NPP	Number of operating WWER-1000 units	Past tHM		Projected tHM											
		on 1/1/98	on 1/1/99	1999	2000	2001	2002	2003	2004	2005	2006	2007	2008	2009	2010
Novovoronezh	1 as of 2005 2 as of 2008 3	212	212	18 6	18 6	18 6	18 6	18 6	18 6	18 6	18 6	18 6	18 6	42 7	42 7
Kalinin	2 as of 2000 3	320	382	47 1	47 1	47 1	47 1	47 1	70 6	70 6	70 6	70 6	70 6	70 6	70 6
Balakovo	4	334	386	94 2	94 2	94 2	94 2	94 2	94 2	94 2	94 2	94 2	94 2	94 2	94 2
South Ukraine	3	404	482	70 6	70 6	70 6	70 6	70 6	70 6	70 6	70 6	70 6	70 6	70 6	70 6
Zaporozh	6	406	542	141 3	141 3	141 3	141 3	141 3	141 3	141 3	141 3	141 3	141 3	141 3	141 3
Rovno	1 as of 2000 2	102	134	23 5	23 5	23 5	23 5	23 5	47 1	47 1	47 1	47 1	47 1	47 1	47 1
Khmelnitski	1 as of 1999 2	102	102	23 5	23 5	23 5	23 5	47 1	47 1	47 1	47 1	47 1	47 1	47 1	47 1
Rostov	as of 1999 1 as of 2004 2	-	-	-	-	-	-	23 5	23 5	23 5	23 5	23 5	47 1	47 1	47 1
Total		1 880	2 240	418 8	418 8	418 8	418 8	465 9	513	513	513	513	536 6	560 7	560 7

## 6. PROSPECTS OF PLUTONIUM TREATMENT

The plutonium accumulated at “PO MAYAK” (30 t of civil plutonium together with 50 t of military plutonium) make the problem of plutonium treatment more complicated. Certainly, measures for reliable plutonium storage should be taken in the near future. Except for the civil plutonium storage, there is a need to store military plutonium. At the same time, due to economical, ecological and political reasons, this storage cannot be effected for a very long period.

The Ministry of Atomic Energy of Russia is permanently studying the possible plutonium, treatment techniques with special attention paid to the problem of civil plutonium extraction and the probable amount of military plutonium. These techniques are based on the following principles:

- A maximum use of Russian experience in plutonium treatment should be made;
- Protection against non-authorized access should be taken into consideration;
- Plutonium treatment techniques should be economically and ecologically acceptable;
- The extracted plutonium treatment technologies should serve as a good basis for the development of an optimum fuel cycle version from a long-term perspective.

The concept of nuclear power centres (NPC), including a reprocessing plant, MOX fuel fabrication plants and plutonium cycle reactors may satisfy such principles. The first centre of this kind may be the centre at “PO MAYAK” near Cheliabinsk, which includes the operating reprocessing plant, RT-1, the MOX fuel fabrication plant (Complex-300) and three planned fast reactors of the type BN-800.

Single cycle burning of MOX fuel in the BN-800 reactors is considered as the first phase of plutonium treatment. Its purpose is early conversion of extracted plutonium into spent fuel to minimize the risk of proliferation. The next phase is recycling and burning of the excess plutonium extracted from spent fuel, including that of MOX fuel. It is believed, that the new core design of the reactors will contain MOX fuel with an increased Pu content and no breeding zones.

The alternative of Pu utilization in thermal reactors is also under considered in Russia. The benefits and disadvantages of plutonium utilization in WWER-1000 reactors compared to those of fast reactors are studied. A critical assembly (SUPR) is being developed at the Physical Energy Institute in Obninsk. It will be used for investigating the safety characteristics of WWERs which use plutonium, including the military plutonium.

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# THE STATUS OF SPENT FUEL TREATMENT IN THE UNITED KINGDOM

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## Abstract

Nuclear power has been used to generate electricity in the UK since the 1950s. Since that time a number of reactor and fuel types have been developed and are currently in use, requiring different spent fuel treatment routes. This paper reviews the spent fuel treatment technology along with the associated waste management and recycle facilities currently in use in the UK.

## 1. BACKGROUND AND GENERAL ISSUES

Nuclear generating capacity in the UK is static with no units currently under construction. The UK's nuclear generating capacity comprises some 8 320 MWe AGRs (advanced gas cooled reactors) and one 1200 MWe PWR operated by British Energy (BE), and 3 350 MWe Magnox operated by British Nuclear Fuels plc (BNFL). The details of the nuclear power stations currently in operation in the UK are given in Table I.

TABLE I. OPERATIONAL NUCLEAR POWER STATIONS IN THE UK

Name	Type	Net Capacity (MW)	Start of Operation	Current Accountancy Lifetime (Years)
Calder Hall	Magnox	200	1956	50
Chapelcross	Magnox	200	1959	50
Bradwell	Magnox	240	1962	↑
Dungeness A	Magnox	440	1965	Average of 37 years operating lifetime
Hinkley Point A	Magnox	460	1965	
Oldbury	Magnox	440	1967	
Sizewell A	Magnox	420	1966	
Wylfa	Magnox	950	1971	↓
Dungeness B	AGR	1140	1983	25
Hartlepool	AGR	1180	1983	25
Heysham 1	AGR	1100	1983	25
Heysham 2	AGR	1240	1988	25
Hinkley Point B	AGR	1170	1976	35
Hunterston B	AGR	1240	1976	35
Torness	AGR	1250	1988	30
Sizewell B	PWR	1200	1994	40

Nuclear power in the UK represents some 18% of installed capacity, but currently supplies some 31% of electricity produced.

Throughout the long history of nuclear power in the UK the dominant form of spent fuel treatment has been reprocessing. Reprocessing has been carried out on a commercial scale in the UK for some 40 years. Initially, the driver for reprocessing in the UK was to separate plutonium for the military programme, but as the civil nuclear power programme began to expand, the driver became more one of resource utilization with the anticipation that the separated plutonium would ultimately be recycled in fast reactors.



The B204 head-end plant was constructed in the late 1960s to allow the Magnox reprocessing facilities to extract plutonium from spent AGR fuel. However, following an accident in the early 1970s, B204 was shut down and never re-opened. Plans were then drawn up for a dedicated plant to reprocess both AGR and overseas LWR fuel. Following a lengthy Public Inquiry in the late 1970s permission was granted for the construction of the Thermal Oxide Reprocessing plant now known as Thorp. Thorp began operation in 1994 and the order book for Thorp is full for the first ten years of operation and is shown in Table II. Reserved Capacity relates to customers who are yet to confirm their acceptance of options.

TABLE II. THORP'S ORDER BOOK FOR THE FIRST TEN YEARS OF OPERATION

Country	Quantity of Fuel (tU)	Fuel Type
UK	2 158	AGR
Germany	969	LWR
Japan	2 673	LWR
Switzerland	422	LWR
Sweden	140	LWR
Spain	145	LWR
Netherlands	53	LWR
Canada	2	LWR
Italy	143	LWR
Reserved Capacity	295	-
Total	7 000	-

Thorp's order book for the second ten years of operation is some 60% full. The total value of Thorp's order book is in excess of £12billion.

The drivers behind the choice of spent fuel management route for the different fuel types within the UK are detailed below. However, it should be noted that Government policy in the UK is that it is for the owners of the spent fuel to decide on the appropriate spent fuel management option based on their own commercial judgement, subject to meeting the necessary regulatory requirements.

## 2. SPENT FUEL TREATMENT

### 2.1. Magnox fuel

Magnox fuel assemblies consist of bars of natural uranium metal, approximately 1m long, which are clad in a magnesium alloy (giving rise to the name Magnox). The Magnox system was designed with a wet discharge route and interim pond storage of fuel in anticipation of early reprocessing. Magnox fuel is reprocessed after about 6 months storage because the cladding does not allow the fuel to be stored for long periods underwater. Although in principle Magnox fuel could be dry stored, the retrofitting of expensive drying facilities or modifications to station fuel discharge routes would prove uneconomic, a fact acknowledged by a UK Government select Committee in 1986, which accepted prompt reprocessing as the only practical option for dealing with Magnox fuel. A Magnox dry store located at the Wylfa reactor site acts as a short term buffer against potential reprocessing throughput constraints, prior to the fuel being ultimately reprocessed.

The reprocessing of Magnox fuel takes place at the B205 facility at Sellafield which has been operational since 1964. This facility, which has a nominal capacity of 1500 t U per year, utilises the PUREX process to separate plutonium and uranium from the waste fission products and actinides. The plutonium and uranium are converted to PuO<sub>2</sub> and UO<sub>3</sub> powders, respectively and stored at Sellafield awaiting utilisation at some point in the future.

The high level waste liquor arising from the reprocessing is stored in cooled tanks prior to vitrification according to a pre-determined programme. The intermediate level waste, principally Magnox cladding, is sent for treatment at the Magnox Encapsulation Plant (MEP) where it is encapsulated in a cement grout and placed in steel drums. Drummed waste are stored in purpose built stores in anticipation of disposal in a deep geological repository.

In 1998, the Magnox power stations, which had been excluded from the privatisation of the nuclear generating industry in 1995, were transferred to BNFL as a wholly owned subsidiary. This means that BNFL is now a uniquely vertically integrated fuel cycle and generation company. The rationale behind the transfer to BNFL being that there is great potential for savings across the whole Magnox fuel cycle.

## **2.2. AGR fuel**

AGR fuel pins are approximately 1m long and consist of enriched  $\text{UO}_2$  pellets clad in a stainless steel tube. The fuel assemblies consist of 36 pins arranged in a circular lattice and sheathed in a graphite sleeve. The AGR power stations have very small at-reactor pond stores, as early reprocessing was envisaged during the design of the reactors, and hence all spent AGR fuel is sent to Sellafield where it is stored underwater. About 700 tU has been reprocessed to date in the Thorp plant. Regarding future AGR spent fuel arisings, the BE contracts for spent fuel management with BNFL (signed in 1995 and 1997) cover arisings over the lifetimes of the AGRs and provide for a near maximum commitment to reprocessing over the first two decades of Thorp operation. Options for further reprocessing following the first 20 years of Thorp operation also exist under the terms of the contracts.

Prior to reprocessing, the AGR fuel assemblies are dismantled in a purpose-built facility at Sellafield and the pins are placed in cans which can then be fed into the shearing facility of Thorp. Thorp, in common with the Magnox reprocessing facilities, utilises the PUREX process to separate out the uranium and plutonium from the waste. The uranium and plutonium are converted within the plant to  $\text{UO}_3$  and  $\text{PuO}_2$  powders respectively and both products are retained within purpose built stores in the plant.

As with the Magnox reprocessing facilities, the high level waste liquor from Thorp reprocessing is stored in cooled tanks prior to vitrification. Intermediate level waste, in the form of cladding and associated components, are encapsulated in a cement grout and sealed in steel drums. Drummed waste is stored in purpose built stores in anticipation of disposal in a deep geological repository.

## **2.3. PWR fuel**

Currently, there is only one PWR in the UK which is Sizewell B. The spent fuel storage pond at Sizewell B was designed to accommodate 18 years spent fuel arisings but has recently been reconfigured to accommodate 30 years spent fuel arisings. BE will consider in due course arrangements for further management of spent PWR fuel in the light of the prevailing commercial and regulatory environment.

## **2.4. SGHWR and WAGR fuel**

Currently, some 160 t U of fuel from the SGHWR and WAGR prototype reactors is being stored at Sellafield. It is expected that all of this fuel will be reprocessed through Thorp.

## **2.5. Fast Reactor fuel**

Following the withdrawal of Government support for the project, the Prototype Fast Reactor (PFR) at Dounreay in Scotland was shut down in March 1994 and is currently being decommissioned.

Support for the European Fast Reactor project (EFR) was also withdrawn, although BNFL is continuing to fund EFR research into plutonium burning (the CAPRA project). Fuel from the PFR has been reprocessed in a mixed oxide reprocessing plant at Dounreay since 1979, with the plutonium arisings transferred to Sellafield for storage. Completion of PFR reprocessing is due around the year 2000 by which time approximately 50 t HM of fuel will have been reprocessed. The UK government has now decided that the Dounreay facilities should close following completion of this and existing MTR reprocessing contracts.

## **2.6. Treatment and disposal of waste streams**

Following the vitrification of the high level waste arising from both Magnox and oxide reprocessing, the glass blocks are encased in stainless steel drums and transferred to a vitrified product store. The vitrified waste is cooled using passive air circulation in the store and it is anticipated that the waste will remain here for at least 50 years before being disposed of in a deep geological repository.

Similarly, the intermediate level waste from both the Magnox and AGR reprocessing are stored following encapsulation in concrete in engineered stores prior to disposal in a deep geological repository (waste arisings from dismantling operations for AGR fuel do not require encapsulation and are stored in steel drums). Solid intermediate waste also arises from the treatment of liquid effluent streams from the operations of the reprocessing plants and this is also encapsulated in cement in steel drums which are stored prior to disposal along with other intermediate level waste.

Solid low level waste is placed in steel drums, compacted and put into half-height ISO containers which are then disposed of at the nearby Drigg facility where they are buried in concrete lined vaults. Liquid low level waste is treated by a number of plants to remove as much of the radioactivity as practicable before being diluted and discharged to the sea.

The overseas reprocessing customers have been offered the choice of receiving back all categories of waste or substituting low and intermediate level waste for a radiologically equivalent quantity of vitrified high level waste. With this option a customer would receive back a single form residue, vitrified high level waste.

## **3. FUTURE DEVELOPMENTS IN REPROCESSING TECHNOLOGY**

With the high commitment to reprocessing in the UK, BNFL is actively investigating improvements in reprocessing technology both in the short term, to optimise the operation of current reprocessing plants, and in the longer term, looking towards the next generation of reprocessing plants.

Shorter term work has concentrated on process optimisation and improvement looking at such measures as reagent consumption, energy usage and waste reduction. In addition, work has gone into enhancing the process envelope to allow a greater range of fuel types and histories to be accepted along with reduction or further treatment of waste streams to respond to the focus on the environmental impact of operations. As a result of the recent OSPAR meeting in July 1998, demanding but achievable targets for the reduction of discharges to the sea have been identified and much work will now focus on providing ways to meet these targets.

For the longer term, novel reprocessing flowsheets are being developed incorporating such ideas as advanced PUREX processes or alternative chemical separation processes. Such processes would be designed to meet the anticipated needs of both the reprocessing customer and other stakeholders towards the third decade of the 21st century.

#### 4. RECYCLE OF URANIUM

Over 15 000 t U of the uranium recovered by Magnox reprocessing has been recycled and about 1650 t U of AGR fuel has been produced from this material. The recycle of reprocessed uranium from Magnox fuel currently has limited strategic benefit as assessed against alternative commercial options. The uranium market conditions are such that further Magnox Depleted Uranium (MDU) recycle is not economic at present. Recycle of the higher residual enrichment product from Thorp is more economic.

Construction of the Line 3 Hex plant at BNFL's Springfield site, for the conversion of reprocessed uranium from Thorp into uranium hexafluoride, is underway with operation due for 2001/2002 and this, along with the newly-opened Oxide Fuels Complex, will allow the recycle of uranium separated in Thorp for both UK and overseas reprocessing customers. The design capacity for Line 3 Hex plant is 1200 t U per annum.

#### 5. RECYCLE OF PLUTONIUM

Recycle of UK plutonium to fast reactors remains the preferred option but, given the withdrawal of Government support for the fast reactor projects, in the shorter term plutonium is continuing to be stored safely and securely under international safeguards at Sellafield. The UK Government policy on the utilisation of plutonium is that it is for the plutonium owner to choose its preferred management option subject to meeting the necessary environmental and regulatory requirements. BE will consider in due course the feasibility of recycling plutonium as MOX fuel at Sizewell B.

#### 6. THE PROSPECTS FOR DIRECT DISPOSAL

As mentioned before, Magnox fuel suffers corrosion problems during long-term storage underwater and hence it is not anticipated that any Magnox fuel will be directly disposed of as this would necessitate expensive changes to the spent fuel management route. AGR fuel, however, does not suffer from this problem and, indeed, BE's current contracts with BNFL allow for some long-term storage as required. It is possible that some such fuel together with PWR fuel may be directly disposed of rather than reprocessed.

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