

# THE SAFETY OF THE NUCLEAR FUEL CYCLE



NUCLEAR ENERGY AGENCY



PARIS



# **THE SAFETY OF THE NUCLEAR FUEL CYCLE**

**NUCLEAR ENERGY AGENCY  
ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT**

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## NUCLEAR ENERGY AGENCY

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- *assessing the contribution of nuclear power to the overall energy supply by keeping under review the technical and economic aspects of nuclear power growth and forecasting demand and supply for the different phases of the nuclear fuel cycle;*
- *developing exchanges of scientific and technical information particularly through participation in common services;*
- *setting up international research and development programmes and joint undertakings.*

*In these and related tasks, NEA works in close collaboration with the International Atomic Energy Agency in Vienna, with which it has concluded a Co-operation Agreement, as well as with other international organisations in the nuclear field.*

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## Foreword

The present report has been prepared by the Working Group on the Safety of the Nuclear Fuel Cycle, which is one of the groups set up by the Committee on the Safety of Nuclear Installations (CSNI) of the Organisation for Economic Co-operation and Development (OECD) Nuclear Energy Agency (NEA). The purpose of the report is to present an up-to-date analysis of the safety aspects in nuclear fuel cycle facilities thus replacing the previous OECD report entitled "Safety of the Nuclear Fuel Cycle", which was published in 1981.

The Working Group commissioned a task force which was charged to prepare this document on the basis of the presently available experience in OECD Member countries.

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This report represents the views of the task force and does not necessarily reflect those of the OECD nor its Member countries. This document is published on the responsibility of the Secretary-General of the OECD.



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# 1. Introduction and Executive Summary

## 1.1 Introduction

This report addresses the safety and technical aspects of fuel cycle operations, and provides information on operating practices. The report is subdivided into the following nine chapters:

- Chapter 1. Introduction and executive summary;
- Chapter 2. The nuclear fuel cycle;
- Chapter 3. Safety principles;
- Chapter 4. General safety topics;
- Chapter 5. Safety issues in the individual stages;
- Chapter 6. Transport of nuclear materials;
- Chapter 7. Safety records of the fuel cycle facilities;
- Chapter 8. Conclusions;
- Chapter 9. Annex: Present and planned facility listing.

Each of the technical chapters can be read separately and is followed by a listing of references. Extensive reference was made to the proceedings of the international conferences held during the 1980's, particularly the Joint American and European Nuclear Societies meeting in 1982, the European Nuclear Society meetings in 1986 and 1990, and the RECOD (REtraitement [Reprocessing], COmbustible [Fuel], DEchets [Waste]) Conference in 1987. Also a selected list of references of the RECOD '91 Conference has been added at the end of Chapter 5.

It should be noted that there are differences between the regulatory policies and/or practices of each OECD Member country, and therefore, only typical examples of these differences are described in this report. It should also be taken into account that nuclear fuel cycle facilities – compared with reactors – are normally operated under both low-pressure and low-temperature conditions and, under sub-critical conditions. Usually, therefore, deviations from normal operating conditions of fuel cycle facilities are less likely to develop rapidly into dangerous situations.

## 1.2 Executive summary

### *Chapter 2. The nuclear fuel cycle*

A general overview is given of the various activities which comprise the nuclear fuel cycle, with the exception of irradiation of nuclear fuel in the nuclear power plant. Radioactive waste management as a whole and its environmental impact are outside the scope of this report, but waste management aspects which are directly connected to fuel cycle facility operation and safety are discussed in detail.

Chapter 2 also describes fundamental principles of the operations, highlights the important safety issues, and reviews the present industrial state-of-the-art. Each phase of the nuclear fuel cycle – starting with uranium mining and milling, followed by enrichment, fuel fabrication, irradiated-fuel storage and fuel reprocessing, waste handling, radioactive material transportation and decommissioning of installations – was prepared with both the non-specialist scientist and the generally educated public in mind.

### *Chapter 3. Safety principles*

This chapter examines the safety philosophy adopted internationally to ensure high standards of safety in the nuclear fuel cycle. After a short review of the basic safety principles, an overview is given of the governmental and regulatory bodies' responsibilities in matters of regulations, guides, licensing procedures and follow-up during operation.

Radiological safety regulations, though specific for each nation, have the common goals of protecting the workers and the public, the confinement of radioactive contamination within safe barriers and the analysis and mitigation of abnormal occurrences. Internal events caused by physicochemical, nuclear, mechanical, natural and human origin are discussed generically.

The topic of safety assessment is reviewed by going through the general stages of identification, analysis and consequence of fault situations by deterministic and probabilistic methods.

The severity scales developed internationally by the Nuclear Energy Agency (NEA) and the International Atomic Energy Agency (IAEA), or nationally (France), are compared and their applicability illustrated. These scales are intended to facilitate greater mutual understanding among the nuclear community, the media and the public, using simple and comprehensible classifications of the nuclear incidents and accidents.

### *Chapter 4. General safety*

Industrial nuclear facilities are subject to internal and external hazards which are adequately designed against. Confinement and ventilation are very important safety requirements which are

generally applicable to all fuel cycle facilities. Criticality is one of the safety hazards associated with the industrial use of fissile materials, against which specific precautions are taken; the factors affecting neutron production and moderation are reviewed in a generic way.

Fire and explosion hazards encountered in laboratories and factories are also identified and preventive measures are discussed on their merits.

Significant progress has been made in the mitigation of the effects of external hazards. The items discussed are: seismic events, explosions and fires, aircraft crashes, extreme weather conditions and floods.

### ***Chapter 5. The safety of the individual stages of the nuclear fuel cycle***

In the front end of the fuel cycle, all activities are focused on fuel fabrication. Uranium mining and milling is the first step of the fuel cycle which involves the handling of large quantities of uranium-bearing ores; at this stage, hazards are primarily associated with the inhalation of uranium-bearing dusts and the potential exposure of workers to Radon (radon-222) and its daughter products, gamma radiation.

Uranium refining and conversion to uranium hexafluoride ( $UF_6$ ) is the ensuing industrial activity. Handling of hexafluoride and gaseous fluorine compounds constitutes the primary source of hazards. Control of these hazards is well understood in the conventional chemical industry.

Enrichment of uranium is carried out industrially by either a diffusion process or a centrifuge process. The release of  $UF_6$  caused by the failure of components in the plant is the main abnormal event which has to be considered. The release of  $UF_6$ , caused by the failure of components in the plants, is one of the abnormal events which has to be taken into consideration. Containment of  $UF_6$  is carefully designed to prevent its accidental release.

Fuel fabrication is the last step of the front end of the fuel cycle by which  $UF_6$  is transformed into uranium dioxide ( $UO_2$ ) subsequently used in the manufacture of fuel elements. For some applications, e.g. Mixed Oxide (MOX) or Fast Breeder Reactor fuel (FBR),  $UO_2$  is mixed with plutonium dioxide ( $PuO_2$ ).

In-plant accidents considered at the design stage are:  $UF_6$  release, criticality and explosion.

The use of reprocessed uranium normally involves only a slightly increased radiological hazard which is taken into account in facility design. The dominant risk in mixed oxide fuel fabrication is release of plutonium to the plant or the surroundings. Special attention is devoted to the discussion of a multi-barrier protection system which protects plant workers and the public from plutonium contamination in both normal and abnormal conditions.

The back end of the fuel cycle starts at the unloading of spent fuel from a nuclear power plant. It includes the transport of spent fuel, storage, reprocessing, radioactive waste treatment and on-site storage of waste. Very extensive experience has been gained in wet storage of spent fuel and only small risks have been identified, e.g. contamination of pool water or leakage of contaminated water.

Abnormal internal occurrences are: loss of electrical power supply, fuel-element handling faults, criticality and loss of coolant; appropriate protective measures are described based on present technology. Internally generated hazards such as loss of pool water or crane collapse also require careful evaluation.

As far as the integrity of storage pools is concerned, there could be a risk caused by internal events and/or external events. The risk is, however, avoided or minimised by appropriate site selection and design of the facility.

Dry storage of cooled spent fuel is also a safe practice. Gradually increasing experience is gained with "away from the reactor" vault type or container storage and independent spent fuel storage facilities at selected reactor sites.

Spent fuel reprocessing is one of the major fuel cycle options. Worldwide experience has been gathered particularly in the OECD countries. Large industrial plants exist or are under construction in France (La Hague), Japan (Rokkasho Mura), and the United Kingdom (Sellafield). In the United States, reprocessing technology for defence purposes is currently operational.

A generic analysis is given of the internal and external hazards specific for reprocessing plants as well as the means to control the risks, among them fire and explosion, which receive the most attention due to the use of industrial quantities of flammable solvents and chemical reagents in the PUREX process.

Reprocessing of UO<sub>2</sub> fuels is discussed in detail – highlighting the hazards during mechanical operations, dissolution of fuel, solvent extraction and waste treatment. The present status of the technology coping with these risks and the means to reduce the hazards are reported. Some recent developments in reprocessing plants (UP3, THORP) are listed in detail.

Specific concerns associated with reprocessing of LWR MOX and FBR MOX-fuel such as criticality and high neutron emission, are discussed. Safety issues related to the production and storage of plutonium are reviewed.

Radioactive waste treatment and storage is a vast subject which is outside the scope of this document. Discussion of related safety aspects has been limited to high level liquid waste (HLLW) storage, waste solidification processes (primarily vitrification), and the storage of vitrified high level waste. The storage of cladding wastes and medium active wastes containing plutonium is also addressed.

The safety issues associated with the storage of HLLW are primarily loss of cooling and containment failure but preventive measures, technical improvements and technical development have considerably reduced the risks associated with this form of storage.

The solidification of high level liquid waste by vitrification is the most widely used technology. The safety problems are generally limited to contamination of equipment and hot cells with a very small external contamination hazard.

Zircaloy cladding, Magnox swarf and insoluble residues are types of high level waste which need to be stored in carefully controlled conditions in order to avoid self-heating reactions which might lead to a resultant pyrophoric hazard. The conditioning method, essentially incorporation into

concrete, is acceptable for interim storage if sufficient provision has been made for release of gases resulting from reaction or radiolysis.

Waste treatment methods include bituminisation of sludges, incineration of plutonium residues and chemical recovery of plutonium from heavily contaminated wastes by acid digestion.

Gaseous discharges from reprocessing (in normal operation) constitute only a small source of radiation dose to individuals.

Decommissioning of nuclear facilities is becoming an increasingly important activity as plants age and become obsolete. Special emphasis is put on those aspects of decommissioning fuel cycle facilities which are by nature and construction quite different from nuclear power plants.

The radiological protection aspects of decommissioning are very important and are currently being addressed by OECD countries.

### ***Chapter 6. Transport of radioactive materials***

Well-established international regulations exist for the transport of radioactive materials, in particular fresh and spent fuel elements, plutonium and waste materials. Stringent safety regulations are based on two main principles: 1) the required safety levels are achieved by appropriate package design, without consideration of the safety of the transport mode; and, 2) the potential hazard defines the required safety level of a package. The packaging of nuclear materials is designed to withstand very severe damage, e.g. collision at 100 km per hour, fire and drop without breach of containment. Specific attention is given to heat dissipation when transporting spent fuel or vitrified waste and to criticality prevention and containment when carrying fissile materials such as plutonium. Transportation by sea and air is receiving more attention than hitherto and is briefly discussed.

### ***Chapter 7. Safety records of fuel cycle facilities***

The experience gained from plant operation constitutes a valuable basis for further safety analyses and design improvements.

In the first section, data are provided on personnel radiation doses in representative fuel cycle facilities during the period 1985-89.

A second section reviews the discharges to the environment in absolute or relative terms compared to the authorisation for each individual fuel cycle facility.

The third section describes, in a summarised form, incidents which have been reported. A series of incidents from the period 1956-80, which were identified in the previous OECD report, and the more recent incidents from the 1980-90 period are included. For each of the described incidents, the technical origin is given and the lessons learned are summarised.

## ***Chapter 8. General conclusions***

This chapter reviews the main issues in the different fuel cycle operations and sketches future considerations for fuel cycle safety.

## ***Chapter 9. Present and planned fuel cycle facility listing in OECD countries***

This listing is based on data from *Nuclear Engineering International* (1989-90), ESARDA and IAEA documents. The data reported were verified by the members of the Working Group for each of the OECD Member countries concerned.

## 2. The Nuclear Fuel Cycle

This chapter provides basic information on fuel cycle facilities – their present status and their interrelations. Individual stages of the fuel cycle are discussed in detail in Chapter 5, with emphasis on each stage's safety record and on the safety aspects involved in its operation. The 1981 Report has been updated using the most recent information. Those processes having reached industrial maturity are emphasized and some processes which do not show great potential have been deleted. Technical information available from some recent conferences has also been included.

### 2.1 Nuclear fuel cycle activities

The nuclear fuel cycle comprises a number of interrelated activities, the possible combination of which provides the various fuel cycle options. The activities comprising the fuel cycle options are listed below:

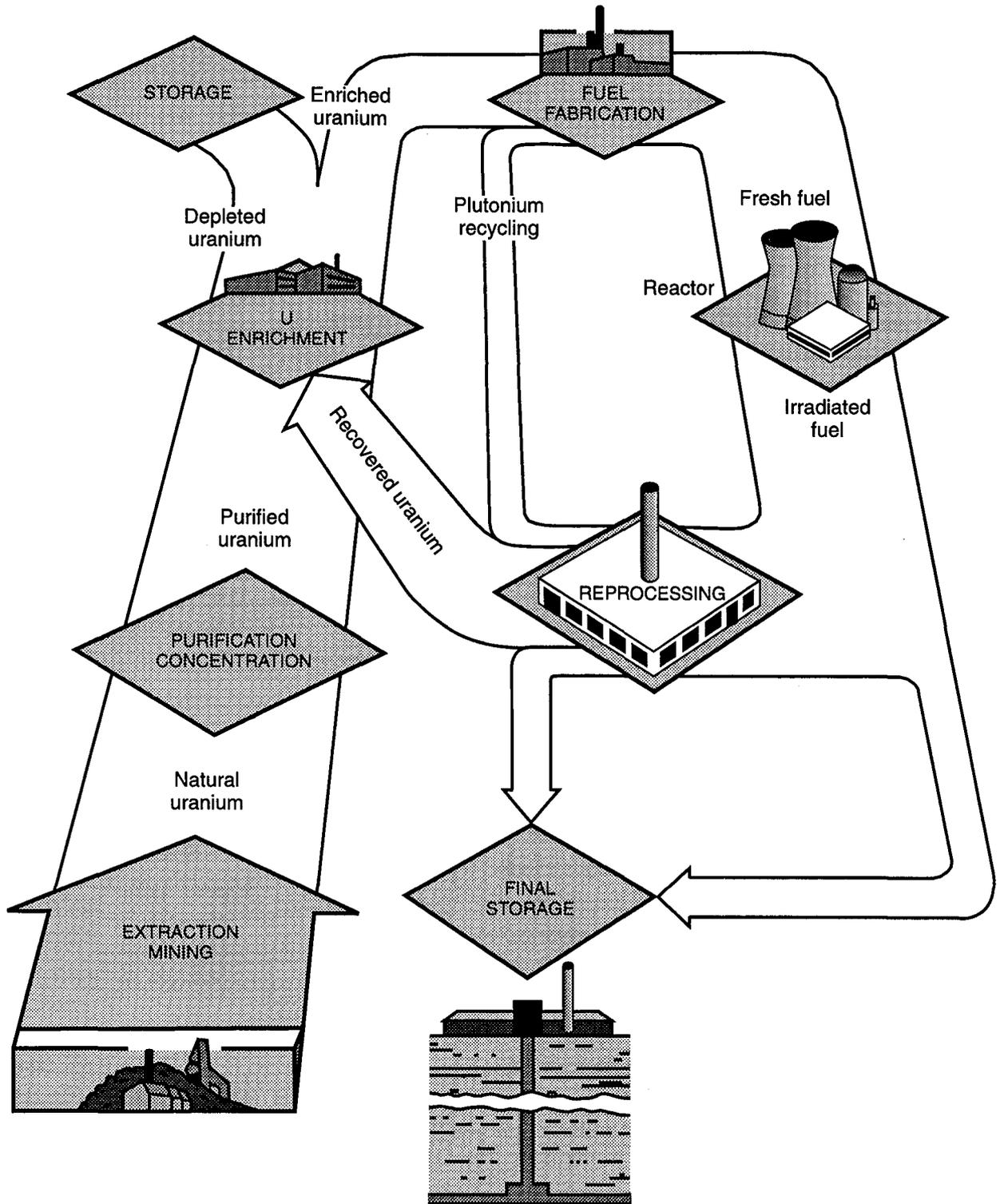
- uranium mining and milling;
- uranium refining and conversion to uranium hexafluoride;
- uranium enrichment and conversion;
- fuel fabrication (including MOX fuel);
- reactor operations;
- spent fuel storage;
- spent fuel reprocessing;
- transportation of nuclear materials;
- decommissioning of nuclear facilities;
- radioactive waste management and disposal options (including spent fuel).

The long-term management of radioactive waste is a very broad field which is not covered in this report. Only the safety aspects associated with processing and storing wastes in the short term are treated in this report.

Disposal of waste is an important issue for all nuclear fuel cycle options. Three main types of fuel cycles are commonly identified, depending on whether or not the spent fuel is reprocessed, and if so, to what type of reactor the extracted plutonium and uranium are recycled. Figure 2.1 shows a diagrammatic representation of the three main fuel cycle options:

- In the once-through fuel cycle, the spent fuel is considered as a waste material and is kept in storage until it is sent for disposal in a geological repository. While this option is presently advocated by Canada, Spain, Sweden and the United States, some other countries are developing it as an alternative. This option does not consider the residual energy resources of the spent fuel and implies the disposal of all long-lived transuranium isotopes (TRU), fission products (FP), and plutonium.

Figure 2.1 The nuclear fuel cycle



- In the thermal reactor cycle the spent fuel is reprocessed and the remaining uranium and generated plutonium are separated from the fission products. The fission products are stored in liquid form for a number of years and then vitrified, to be stored for several decades in engineered storage facilities. Ultimately the vitrified high level waste and the TRU waste will have to be disposed of. This option is followed by Belgium, France, Germany, Japan, Switzerland and the United Kingdom. Some of these countries already recycle plutonium and uranium in thermal reactors.
- In the fast breeder reactor recycle, spent fuel from both the highly enriched core and the surrounding blanket region is reprocessed and the resulting uranium-plutonium product can be recycled for use in a fast breeder reactor. Recycling of these fuel types has been shown to be feasible on an industrial prototype scale. The expected growth of the fast breeder option did not materialise as expected mainly because of economic reasons. This option has been mainly developed in France, Japan, the United Kingdom and some other countries.

A combination of both LWR and FBR fuel cycles can provide an effective use of depleted and recycled uranium and plutonium resulting from subsequent reprocessing cycles should there be a sufficient rise in uranium prices in the future.

In both the thermal and fast breeder reactor cycles, facilities for the storage of separated plutonium have to be installed if an imbalance occurs between the reprocessing plutonium output and the fuel fabrication requirements for thermal and fast breeder reactors.

Plutonium stored over a relatively long period of time becomes contaminated with the decay product, americium, which interferes with the normal fuel fabrication procedures.

Transport of plutonium in either solid or solution form from the reprocessing plant to the fuel fabrication plant, which are not generally co-located, requires careful consideration.

The choice of a specific fuel cycle option depends on a variety of different and sometimes interrelated factors. The economics of nuclear electricity production, the availability and price of uranium on the world market, the availability of other energy sources, safety factors, environmental considerations as well as policies and political issues all play a role in the final choice.

Until about 2005, uranium requirements, spent fuel production, reprocessing capacity, enrichment capacity and fuel fabrication capacity can be reliably predicted on the basis of the present operating and scheduled nuclear power capacity. Beyond that date it becomes difficult to forecast with any precision the future significance of, for example, the fast breeder option and other secondary nuclear power plant options. Even the projected rate of LWR nuclear power growth in that later period is very uncertain.

This report primarily considers LWR reactors fuelled with uranium oxide but also discusses to an extent the growing option of MOX recycling in thermal reactors. Fast breeder reactors and gas-cooled reactors are taken into account, but their overall significance is decreasing in the global nuclear picture compared to the expectations of some years ago.

## 2.2 Status of the various fuel cycle segments

In this section the various segments of the fuel cycle are briefly described with emphasis on the present state of development and new ventures.

### 2.2.1 Uranium mining and milling

Uranium is widely distributed in nature and is found in concentrations which make it economically feasible to mine. Mining operations to recover uranium from ore, as low as 0.1 per cent uranium, have been viable. Significant ore bodies are being developed in Canada which have uranium concentrations of up to 15 per cent.

In many countries uranium has been recovered as a by-product from other mining activities or from chemical processing industry feed streams, such as phosphoric acid in fertilizer plants.

The total known resources of uranium for selected countries [excluding the CIS, most East European countries and China] are estimated at about  $3.1 \times 10^6$  tonnes. The present world operational capacity for extracting uranium from ore amounts to about 50 kT per year (1). Although production has been less than demand since 1985, there is currently a surplus of uranium, and many facilities are consequently shut down.

Beyond the end of this century, supply and demand should be more in balance. Assuming all planned and prospective production facilities are developed, no serious production shortfalls are expected. Excess uranium stock piles from Russia and other former Soviet Republics are likely to have a major impact on supply and demand.

A typical process flowsheet for the extraction of uranium is shown in Figure 2.2. Typically this process comprises crushing and grinding of the ore followed by chemical leaching with acid or alkaline solutions. The acid leaching process is more widely employed, usually using sulphuric acid because of its relatively low cost and more acceptable environmental impact as compared with other acids. The alkaline leach process is used for ores which would require excessive amounts of acid. An advantage with using an alkaline carbonate leach process is that it is rather selective for uranium, thus leaving much of the radium out of solution.

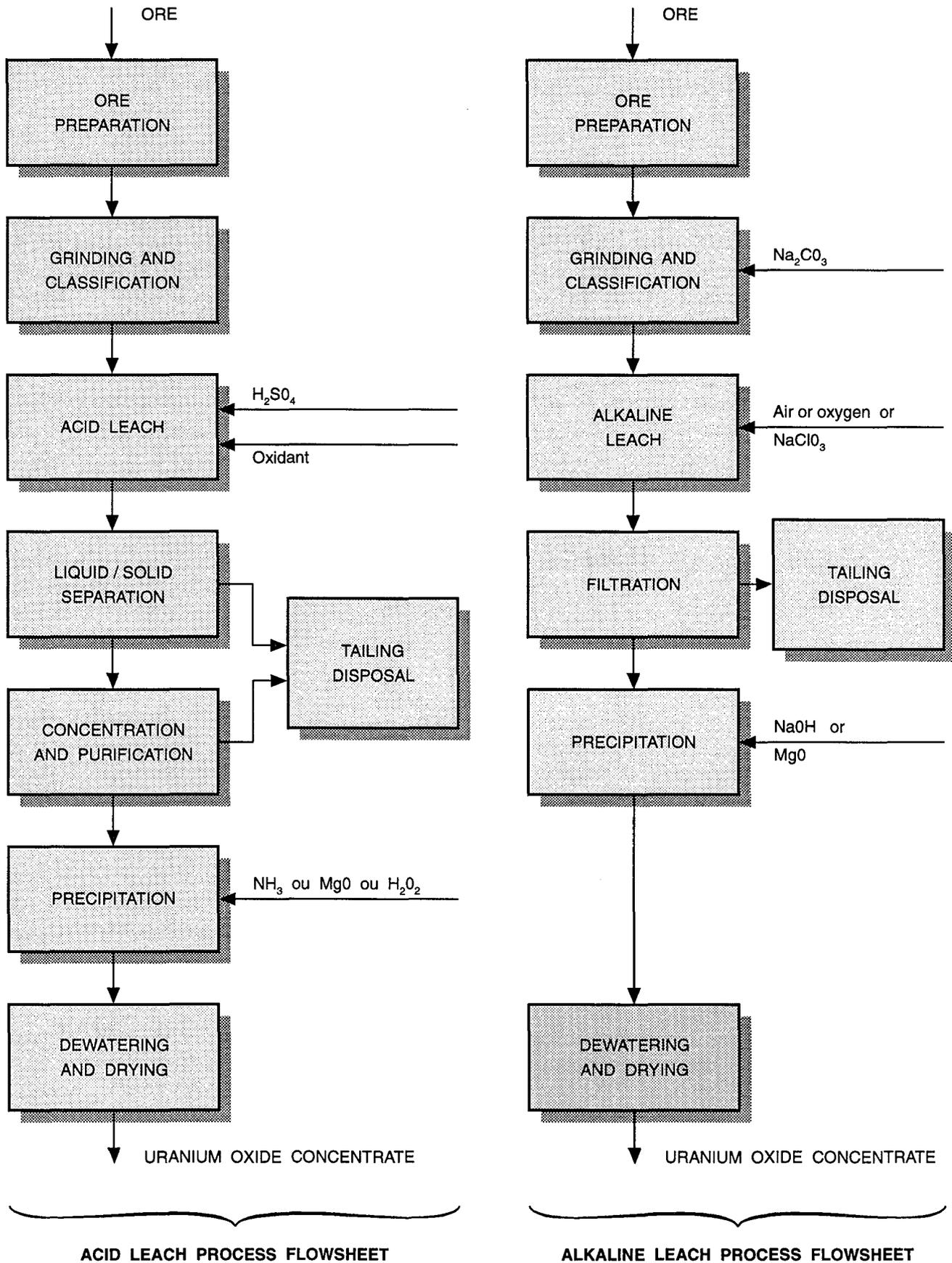
- Sulphuric Acid Leach

This process comprises crushing and grinding of the ore, and then leaching with sulphuric acid and an oxidizer, such as hydrogen peroxide. The resultant slurry is filtered and clarified, and transferred to an ion exchange unit-operation which produces a uranium-bearing solution. The uranium is precipitated from the solution in the form of one of the diuranates, which is dried to produce a uranium oxide concentrate ( $U_3O_8$ ) which is further purified in a refining facility.

- Alkali Carbonate Leach

This process comprises crushing and grinding of the ore, and then leaching with a sodium carbonate solution and an oxidizing agent such as sodium chlorate, heated and under pressure to dissolve the uranium. The leach liquor is separated from the solids by filtration, and the uranium is precipitated with an alkali, usually magnesia or caustic soda, as a sodium or magnesium diuranate dried to produce a uranium oxide concentrate ( $U_3O_8$ ) which is further purified in a refining facility.

Figure 2.2 Uranium Extraction from Ores



The term "yellow cake" has been used in a generic fashion to describe the family of uranium precipitates, the so-called "diuranates", and to describe impure uranium oxide. In fact, the term "yellow cake" should only be applied to ammonium diuranate.

It is forecast that the accumulation of uranium mill wastes (tailings) will exceed  $10^9$  tonnes by the year 2000, corresponding to annual uranium requirements at that time of the order of  $10^5$  tonnes of uranium.

The mining and milling of uranium ores are not activities which would give rise to incidents that could have serious radiological consequences for the general public or the environment. Thus these facilities are viewed as conventional mining industry activities not requiring comprehensive analyses of radiological accidents. Their off-site impacts have not, until recently, attracted the public concern generated by other nuclear facilities. Uranium mining in underground mines (versus open-pit mining methods) has attendant radiological implications for workers (3, 4, 60). Hence, the health physics community is paying greater attention to radiation protection practices to ensure that the combined effects of radon daughters, radioactive dust and external radiation are adequately addressed. The recently discovered high-grade ore bodies which are being developed as underground mines will require that remote mining methods be developed and that much more stringent radiation protection procedures be implemented.

### ***2.2.2 Uranium refining and conversion to uranium hexafluoride***

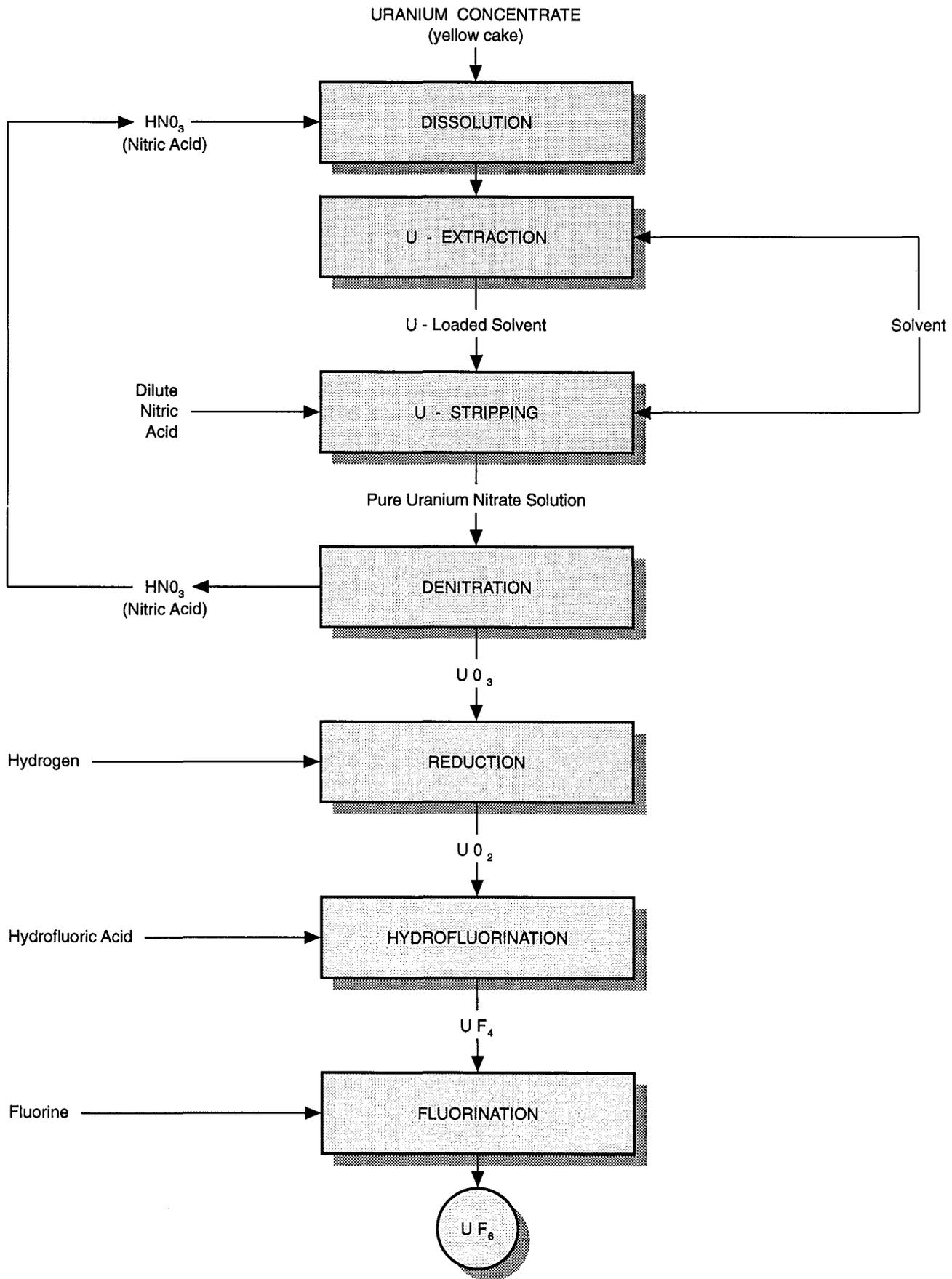
The end product from the previous stage of the fuel cycle is sometimes called "yellow cake" which consists of impure  $U_3O_8$ . Refining or purification processes are required to bring the uranium oxide compound to a nuclear grade purity before it is converted through a sequence of chemical forms ( $UO_3$ ,  $UO_2$ ,  $UF_4$ ) to  $UF_6$  or U metal. Figure 2.3 shows a typical flow sheet of the processing sequence.

The uranium refining and conversion capacity in the OECD countries amounts to about 52 kT U per year while current conversion requirements are approximately 43 kT U per year and is situated primarily in Canada, the United States, France and the United Kingdom. The bulk is transformed from  $U_3O_8$  to  $UF_6$  for subsequent enrichment, and only about 10 per cent remains in the form of  $UO_2$ , or converted to U, for use in heavy water-moderated reactors or graphite-moderated and material-testing reactors.

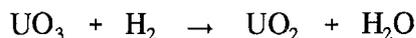
The  $U_3O_8$  received by the uranium refineries is dissolved in nitric acid. The resulting solution of  $UO_2(NO_3)_2 \cdot 6H_2O$  (UNH) is fed to a solvent extraction operation generally consisting of either pulsed columns or mixer settlers contacting the aqueous UNH solution counter-currently with an organic extractant, generally tributyl phosphate (TBP) diluted with kerosene or dodecane.

The uranium product, complexed by TBP, is back extracted by a dilute nitric acid solution and concentrated by evaporation until a solution density of about 400 to 1,000 grams per litre is obtained.

Figure 2.3 Uranium Refining and Conversion to Hexafluoride

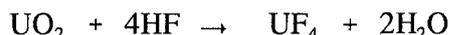


This solution is calcined to pure  $\text{UO}_3$  in a fluidised bed or batch-pot contactor. Alternatively, the concentrated UNH is neutralised with gaseous ammonia and then calcined to  $\text{UO}_3$ . The resulting  $\text{UO}_3$  is reduced to the  $\text{UO}_2$  form by  $\text{H}_2$  in fluidised bed reactors according to the reaction:

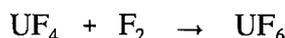


Some of the uranium is directly used as  $\text{UO}_2$ , e.g. in the CANDU reactor.

The next step in the uranium processing is its transformation into  $\text{UF}_4$  by reacting it with anhydrous HF according to the reaction:



This reaction is carried out at the temperature range of 300 to 500°C in fluidised bed reactors or rotary kilns. This chemical reaction also takes place in a  $\text{UO}_2/\text{H}_2\text{O}/\text{HF}$  slurry. The solid  $\text{UF}_4$  powder is transformed into the gaseous  $\text{UF}_6$  by reacting it with pure fluorine gas at 500°C according to the reaction:



Off-gas purification steps including fluorine removal are necessary operations to ensure environmental protection.

The  $\text{UF}_6$  product leaving the fluorination reactor is condensed at  $-10^\circ\text{C}$  and placed into cylinders which are shipped to enrichment facilities.

The transformation of  $\text{UO}_2$  into  $\text{UF}_4$  with HF, and particularly the fluorination of  $\text{UF}_4$  with pure  $\text{F}_2$ , involves the use of corrosive and reactive reagents which react with all oxygen-containing substances. Special care is taken to avoid leaks in the chemical reactors which are used to perform these conversion reactions, and to avoid releases within the facility and to the outside environment. In many cases the toxicity of these conventional chemicals is more of a concern than the radiological aspects of processing uranium.

The use of flammable reagents in industrial quantities is controlled to minimise the risk of fire and explosion which would cause an uncontrolled release of relatively large quantities of uranium.

### **2.2.3 Uranium enrichment**

Light water reactors represent about 90 per cent of the nuclear power reactors. They use enriched uranium as fuel, i.e. uranium in which the percentage of the fissile uranium isotope – uranium-235 – has been increased from its natural state of 0.7 per cent to about 5 per cent, the balance being uranium-238. The remaining fissile uranium in the tail end product varies between 0.2 and 0.3 per cent and is determined by technical and economic factors.

Enrichment is based on a physical process by which lighter isotopes are separated from heavier ones, e.g. uranium-235 from uranium-238. The industrial-scale processes are either gaseous diffusion through membranes or centrifugal acceleration in high-speed centrifuges.  $UF_6$  gas passes through cascades of enrichment unit steps to reach the desired degree of enrichment in the lighter isotopic component.

The Separative Work Unit (SWU) is a measure of the uranium throughput of an enrichment plant and of the separation work to be carried out in order to enrich uranium. It varies according to the concentration of uranium-235 in feed, product and tails. For example, to produce one kilo of enriched uranium at 3.5 per cent requires 6 to 8 kilos of natural uranium at 0.7 per cent of uranium-235 and consumes 5 SWU (leaving a tail uranium-235 concentration of 0.2 to 0.3 per cent).

The world capacity amounts to 33,000 SWUs (or, 33 MSWU) per year of which 30 MSWUs are based on the gaseous diffusion process and three MSWUs on the centrifuge process. The annual enrichment requirements for the OECD countries amount to 23.4 MSWUs which show the extent of overcapacity existing in the world.

The large uranium enrichment plants based on the gaseous diffusion process, are situated in France and the United States. The centrifuge enrichment plants are situated in Holland, Germany, the United Kingdom and Japan. Smaller plants based on both types of processes are operating in a number of countries.

The isotope diffusion process is highly reliable although quite energy intensive and, therefore, more expensive than the centrifuge process.

The EURODIF plant at Tricastin has operated for about ten years (6, 7) during which it has produced 60 MSWUs. The plant operates 365 days per year on a 24-hour basis.

The URENCO centrifuge plants (in Germany, the Netherlands, the United Kingdom) with an overall capacity of 2.6 MSWUs per year have developed steadily because of their economic advantage compared to gaseous diffusion, and the technical maturity of the centrifuge technology (8). Centrifuge failures of less than 1 per cent per year have been obtained industrially over a 13-year period totalling several million machine years of operating experience without incidents (9).

The Japanese pilot plant at Ningyo-Toge was shut down in 1990 after more than ten years of operation. A demonstration plant with an overall capacity of 0.35 MSWUs per year started operation in 1988. A commercial plant at Rokkasho is now partially operational and is expected to reach a capacity of 1.7 MSWU per year by the year 2000.

A specific area of expansion for gas centrifuge-based enrichment capacity exists in its use for the recycling of reprocessed uranium. Enrichment by both diffusion or centrifuge techniques has been demonstrated to be a safe and reliable procedure. No major radiological incidents have been reported.

New enrichment processes are at the pilot and/or demonstration stage; these include the laser enrichment process (10, 11, 12) and the chemical enrichment process.

In Japan, the ASAHI chemical enrichment plant observed higher efficiencies in pilot-plant operation than on bench scale. This plant was terminated in 1992.

The occurrence of increasing amounts of uranium-232, uranium-233, uranium-234 and uranium-236 in recycled uranium has an influence on its neutronic behaviour and its radiological impact. Additional enrichment will increase the uranium-236 isotope concentration which has a neutron capture cross section high enough to marginally decrease the overall reactivity of recycled uranium. The concentration of uranium-236 may build up from nearly zero to 0.42 per cent at 33.2 GWd/T, and up to 0.64 per cent at 50.5 GWd/T (14).

The radiological impact concern results from the increased uranium-234 activity and from the presence of uranium-232 which is a parent isotope of the 4n natural series producing the gamma emitters thorium-228 and thallium-208. After a number of years this radionuclide will build up in recycled UF<sub>6</sub>.

Results based on the industrial-scale operations with reprocessed UF<sub>6</sub> have been reported by URENCO (15). Some neptunium but very little plutonium and fission products were found in the UF<sub>6</sub> feed from reprocessed fuel. The centrifuge enrichment process is very well-suited to handle these relatively low contamination levels due to the small in-process inventory of the centrifugation cascade and the relatively maintenance-free operation.

Procedures to deal with the presence of technetium-99 might be required when recycled uranium is introduced into an enrichment plant (15).

Two processes based on laser excitation of atomic uranium vapour or of UF<sub>6</sub> molecules have been studied. The most attractive is the Atomic Vapour Laser Isotopic Separation (AVLIS – [SILVA in France]) process which relies on the selective ionization of uranium-235 vapour by a coherent light beam emitted by a powerful copper vapour laser which energises a dye laser tuned precisely to one of the wavelengths of uranium-235 vapour. The ionized uranium-235 atom is extracted from the atomic vapour cloud by electromagnetic field separation and condenses as enriched uranium metal. A single stage is sufficient to obtain 3 to 4 per cent enrichment.

The Molecular Vapour Laser Isotopic Separation (MVLIS) process relies on a difference between the excitation energies of <sup>235</sup>UF<sub>6</sub> as compared to <sup>238</sup>UF<sub>6</sub>. Taking into account the enormous complexity of the vibrational and rotational states of the UF<sub>6</sub> molecules this technique – though in principle less sophisticated – does not lead to good established selectivity because of the overlapping of band structures of both isotopic molecules.

#### 2.2.4 Fuel manufacture

Nuclear power reactor fuel can be uranium metal, uranium oxide or uranium-plutonium oxide called mixed oxide (MOX).

The production of U metal (natural or slightly enriched) continues at a level of about 2,400 THM per year and this fuel is used in gas-cooled reactors in France and the Magnox reactors in the United Kingdom. When these reactors close down at the end of their lifetime, this type of fuel fabrication will be phased out.

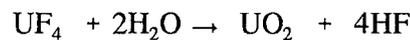
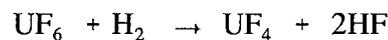
Uranium oxide fuels serve the main reactor programmes. Production has reached industrial capacities of 8,300 THM per year for LWR fuel and 2,200 THM per year for CANDU fuel in the

OECD countries. The annual  $\text{UO}_2$  fuel requirements are about 7,500 THM. Recycling of plutonium in power reactors has become a new industrial venture since the MOX production facilities for fast reactor fuel have partially (60 T MOX fuel per year) been reconverted for production of thermal MOX fuel. The production capacity of MOX fuel for breeder reactors has been reduced since there is little likelihood for a considerable expansion of this reactor type before the end of the 1990s. France and Japan are producing 20 TMH and 5 TMH fuel per year, respectively.

#### 2.2.4.1 Uranium oxide fuel (enriched uranium)

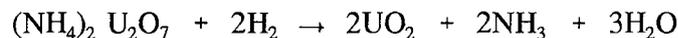
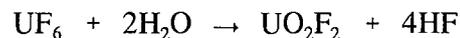
Enriched depleted uranium are usually stored as  $\text{UF}_6$ . In order to produce  $\text{UO}_2$  fuel, it is necessary to reconvert the  $\text{UF}_6$  into  $\text{UO}_2$ . Three processes have been used to carry this out (see Figure 2.4):

- Reduction of  $\text{UF}_6$  to  $\text{UF}_4$  with hydrogen followed by hydrolysis of  $\text{UF}_4$  with steam according to the reactions:

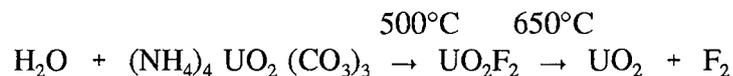
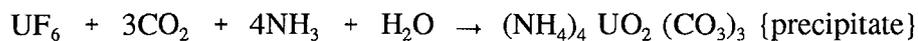


These occur in a single integrated kiln – the Integrated Dry Route (IDR)

- Direct conversion of  $\text{UF}_6$  into  $\text{UO}_2\text{F}_2$  followed by precipitation with ammonia to form ammonium diuranate (ADU) and reduction with  $\text{H}_2$  to  $\text{UO}_2$ :



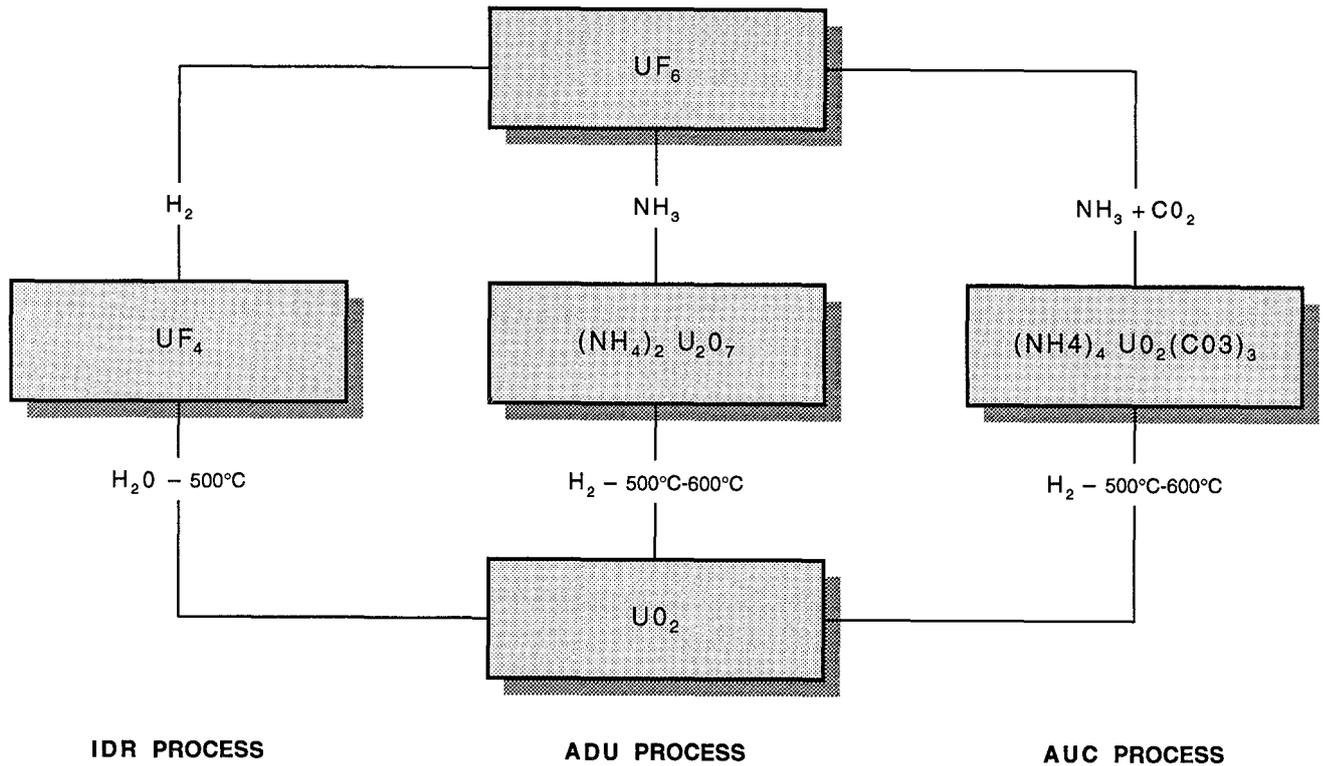
- The AUC process by which  $\text{UF}_6$  is transformed into Ammonium Uranyl Carbonate (AUC) by treatment with  $\text{CO}_2$  and  $\text{NH}_3$  in water:



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

No major incidents with radiological consequences have been reported, but some incidents (mainly dealing with the transfer of  $\text{UF}_6$  to the plant) have been reported at the conversion plants, in France and Germany.

Figure 2.4  $UF_6$  to  $UO_2$  conversion processes



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

With the entry of reprocessed uranium into the fuel fabrication plant some complications may arise due to the presence of uranium-232, additional uranium-234 and uranium-236 which are more radioactive than the natural isotopes (uranium-235, uranium-238). The presence of this wide spectrum of uranium isotopes in facilities which were not originally designed for that purpose requires that the operations be adequately assessed. Each of the fuel fabrication steps produces "Clean Reject Oxides" (CRO) and "Dirty Reject Oxides" (DRO) which have to be recycled or considered as waste materials.

#### 2.2.4.2 Mixed oxide fuel (MOX)

Historically the major need to develop MOX fuel came from the breeder reactor technology since it was assumed that FBRs would take advantage of the plutonium recovered from the reprocessing of fuel. Mainly for economical reasons, the FBR has not been utilised as much as expected.

Reconversion of MOX fuel fabrication plants from FBR fuel to LWR fuel has taken place recently. An overview of the role of plutonium recycled in LWRs can be found in (62). Fabrication of MOX fuel usually starts with the mechanical mixing of uranium and plutonium oxide powders or by "diluting" a master blend of  $UO_2$ - $PuO_2$  with  $UO_2$  to the required enrichment. The MOX fuel

fabrication process (MIMAS) developed by BN (Belgonuclaire) and COGEMA is shown in Figure 2.5 (16).

In the United Kingdom, BNFL developed a binderless route for the direct blending of  $\text{PuO}_2$  and  $\text{UO}_2$  powders to the correct MOX powder enrichment, thus eliminating the requirement to produce a "master blend" for dilution with  $\text{UO}_2$ . In this route, very homogeneous MOX powders are produced in a short time by blending the  $\text{PuO}_2$  and  $\text{UO}_2$  powders in a high-energy attritor mill. This process is being used by BNFL in collaboration with AEA Technology in the MOX Demonstration Facility – an 8 THM MOX plant due to start production of MOX fuel in early 1993.

The milling of  $\text{UO}_2$  and  $\text{PuO}_2$  powders, often called micronisation, is an important part of the mechanical mixing process to ensure that a solid solution of U-Pu $\text{O}_2$  is produced during sintering of the MOX pellets. The formation of a solid solution of U-Pu $\text{O}_2$  is important to ensure solubility of the pelleted oxide in nitric acid for reprocessing.

The fabrication of MOX fuels results in the generation of plutonium-contaminated waste materials and scrap fuel residues in the form of CROs and DROs. The recycle of these scrap fuel residues to maximise production plant efficiency and minimise the costs of both fuel production and waste management, is an important part of fuel production plant management.

The conversion of  $\text{PuO}_2$  separated from LWR fuels into MOX gives rise to the need for strict operator dose control, effective shielding and, in some cases, remote-handling techniques to be adopted. This situation arises because of the neutron dose emitted from the plutonium-238 and plutonium-240 isotopes present in the plutonium and the gamma dose emitted from the americium-241 which may have "grown into" the oxide by the decay of plutonium-241 during storage. Security measures associated with MOX production are very important.

An alternative route to the production of MOX fuel is based on co-precipitation and is known as the AUPuC process developed by ALKEM (Germany) (17). This is similar to the AUC method but the uranium and plutonium are precipitated simultaneously as  $(\text{NH}_4)_4(\text{U-Pu})\text{O}_2(\text{CO}_3)_3$  in aqueous solution. The precipitate is a completely homogeneous solid solution which, after calcination at  $600^\circ\text{C}$ , produces a free-flowing powder of U-Pu $\text{O}_2$ , product which is soluble in boiling  $\text{HNO}_3$ . However, this process requires the transportation of plutonium in aqueous solution if the MOX plant is not situated near the reprocessing plant. The advantage of the method lies in the absence of dust throughout the production process but its drawback lies in the criticality limitation inherent to any aqueous plutonium processing method. A third process for the production of  $\text{UO}_2$ -Pu $\text{O}_2$  is still closer to reprocessing technology. A co-conversion of uranium-plutonium nitrate has been developed by PNC (Japan) (18) and is based on dehydration and denitration of product solutions from the reprocessing plant by microwave heating.

The flowsheet of the process is shown in Figure 2.6 and constitutes a mid-way between the full dry-powder process of BN-COGEMA and the process aqueous precipitate of AUPuC. The process requires a very efficient off-gas treatment unit but lends itself very well to automation and operation in shielded gloveboxes. Since crushing and milling operations of  $\text{UO}_2$ /Pu $\text{O}_2$  are incorporated in the process, dust containment is a major concern.

Figure 2.5 Typical MOX Fuel Fabrication Process

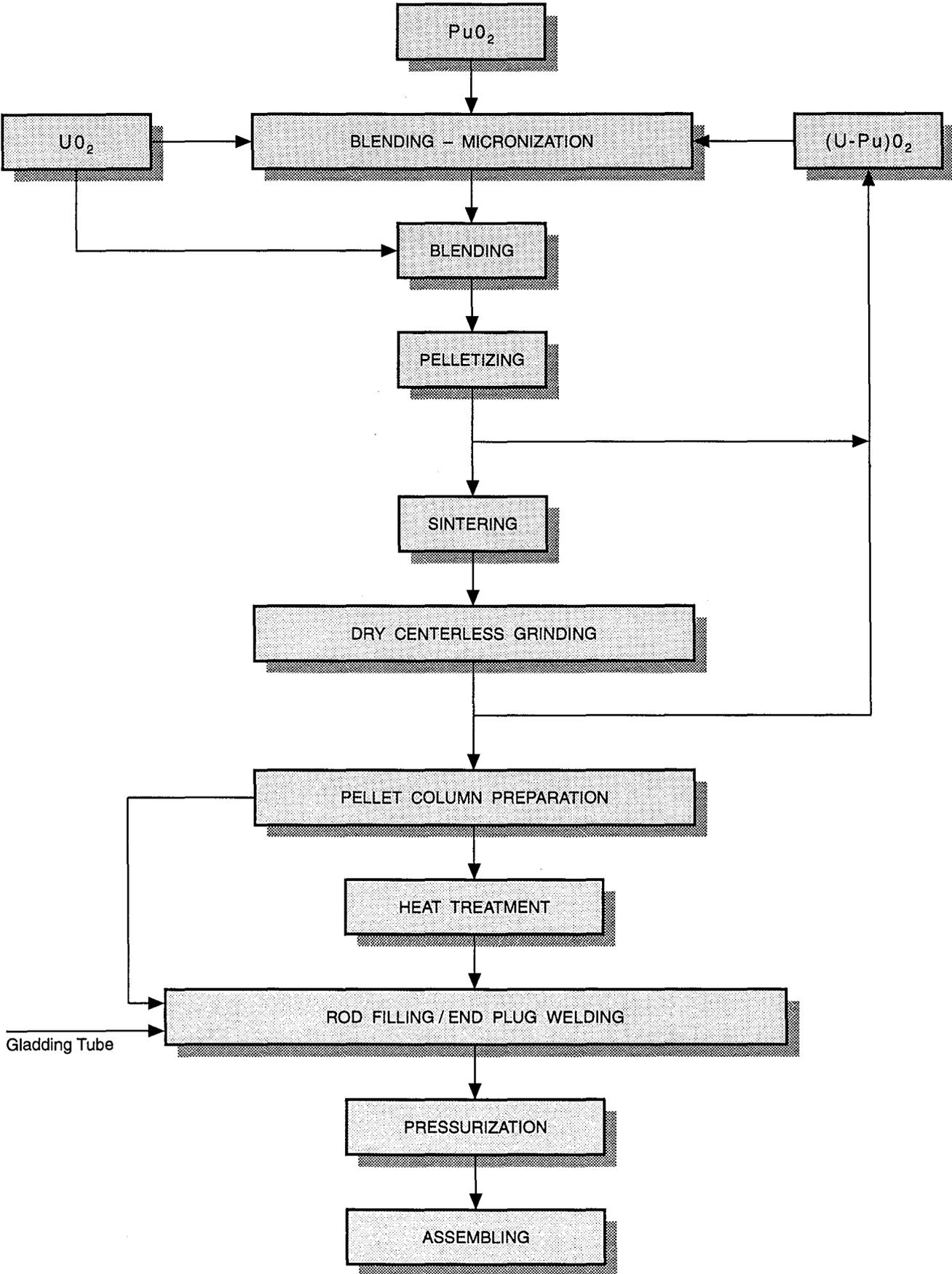
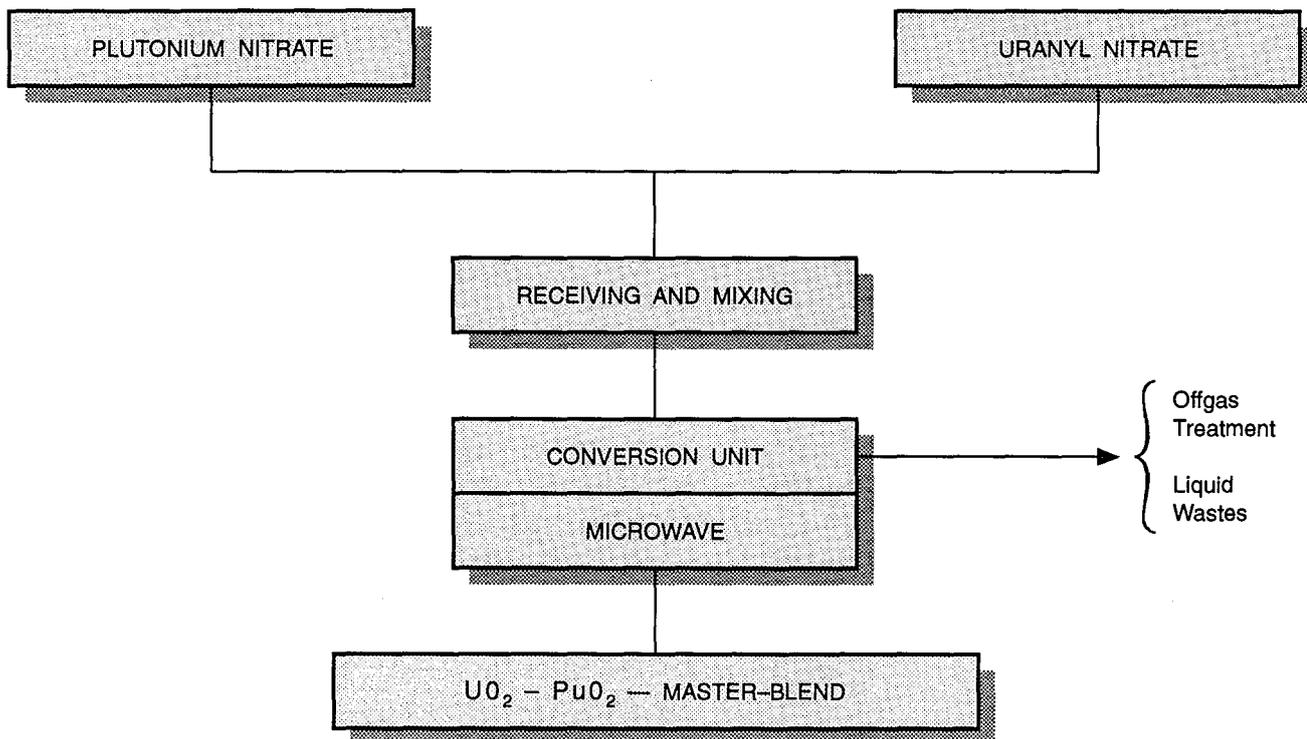


Figure 2.6 Direct production of MOX powders from uranium-plutonium-nitrate solutions



### 2.2.4.3 Fast reactor fuel

Extensive pilot-scale experience has been gained in industrial facilities for the production of U-PuO<sub>2</sub> with 15 to 30 per cent Pu content. The difference between LWR and FBR fuel lies in the Pu concentration and in the type of cladding (stainless steel or ferritic steel) which is used. The criticality problems involved in producing industrial quantities of uranium-plutonium (20 per cent) O<sub>2</sub> fuel have been solved.

In the United Kingdom, the BNFL Sellafield plant has an annual capacity of 5 T MOX-FBR for use in the Dounreay PFR. In Germany, Alkem's Hanau plant has a yearly capacity of 10 T MOX-FBR for KNKII and SNR300. In Belgium, the BN plant has an annual output of 8 T MOX-FBR and produced fuel for KNKII and SNR300. In France, CEA's Cadarache plant has produced 100 T of MOX since 1964, particularly for the prototype Phenix reactor (4 T per year) and the power plant Superphenix (20 T per year). In Japan, a new MOX fuel fabrication plant at Tokai with a 5 THM per year MOX-FBR output has been operated in order to provide the necessary fuel for the prototype fast breeder reactor, Monju.

Uranium-plutonium carbides and nitrides have been proposed and studied as alternatives to U-PuO<sub>2</sub>, but this option has not been developed beyond the bench scale.

#### 2.2.4.4 *Metal fuel*

Metallic uranium was the first type of fuel developed and is still used in high-flux reactors and gas-cooled reactors. Uranium dioxide (UO<sub>2</sub>) is converted to UF<sub>4</sub> and this compound is mixed with magnesium or calcium shavings and pressed into pellets. By heating the pellets, the reaction:



produces U metal which is separated from the MgF<sub>2</sub> or CaF<sub>2</sub> slag.

Uranium metal alloyed with aluminium is used in high-flux material test reactors.

In UO<sub>2</sub> fuel fabrication plants protection from releases of radioactive compounds is assured by dynamic barriers such as ventilation hoods and engineered safety systems.

Mixed oxide fuel fabrication facilities have at least two physical barriers separating plutonium from the ambient environment: the first physical barrier is the glove box and the second is the containment building. In addition, a dynamic barrier (ventilation) protects the workers, and high-efficiency particulate filters (HEPA) confine releases within the facilities. Fire, explosions, criticalities, and external events are the potential causes, leading to loss of containment, against which adequate safety preventive measures are taken.

#### 2.2.5 *Spent fuel storage*

The nuclear electricity production in LWRs produces about 27 THM/GWe per year, at typically 33,000 MWd/T, as spent fuel. This fuel is initially stored at the reactor site in pools. The fuel is later transported to a reprocessing plant equipped with large storage ponds (La Hague, Sellafield), away from reactor storage pools, a facility awaiting further processing (delayed reprocessing), or disposal site.

The cumulative spent fuel arisings in the OECD countries amounts to approximately 84,000 THM and is expected to grow to approximately 174,000 THM by the year 2000. The available storage capacity on the reactor sites and away from the reactor sites amounts to approximately 150,000 THM in 1990 and is expected to increase to 200,000 THM in the year 2000.

The total installed pool storage capacity away from the reactor sites is about 30,000 THM of which 23,000 THM capacity is situated at the reprocessing sites of Sellafield and La Hague. The residual 7,000 THM is dispersed throughout the world except for two larger away from reactor facilities independent of the reprocessing plants, situated in Sweden (CLAB 5,000 THM) and Finland (Olkiluoto-Loviisa, 1,270 THM).

The storage of spent fuel requires: shielding to handle the fuel; cooling to maintain fuel at a safe temperature; criticality safety control; means and facilities for control of water chemistry; and, decontamination.

Dry storage capacity has been constructed for oxide at Gorleben (1,500 THM) and Ahaus (1,500 THM) in Germany, and at Wylfa in the United Kingdom for Magnox fuel.

The Monitored Retrievable Storage (MRS) facility of 15,000 THM (to be built in the United States) has not as yet been sited.

#### 2.2.5.1 *Pool storage*

Pool storage capacity (~ 120,000 THM) at reactor sites is by far the most important but a detailed discussion on it is not within the scope of this report. It is, however, important to draw attention to an extensive study carried out by the IAEA and OECD/NEA on the storage of LWR spent fuel in pools (20). This survey concluded that no degradation of stored LWR fuel had been observed over a time period of up to 23 years. In the meantime another decade has been added to this good safety record of pool storage. The document recommends that fuel examination should be ongoing if spent fuel storage in pools continues for several decades. In order to cope with an increasing inventory of spent fuel, utilities in the United States and other countries have undertaken actions to increase the in-pool capacity by reracking and the addition of a second row of racks (21). Reracking involves the change of the rack configuration by which smaller distances between individual assemblies are obtained. This practice requires the addition of more neutron absorbing material, adequate cooling, and design against earthquakes. Improved neutron absorbers based on boron carbide containing materials have been developed and successfully installed.

Experimentally, rod consolidation has been investigated and involved mechanically removing the fuel rods from the fuel assembly hardware, and placement into a canister. Effective consolidation also requires the compaction and disposal of the grid spacers, guide tubes and end fittings.

#### 2.2.5.2 *Dry storage*

Dry storage has been developed but is not yet used to the extent of wet storage, and it is probable that for safety, technical, and economic reasons, commercial fuel will always need a cooling period in wet storage. However with the increasing storage period after discharge from the reactor, dry storage may have advantages over wet storage. From a technical and design point of view, the transport and dry storage of spent fuel are very closely linked to each other by the technical requirements imposed on transport storage flasks.

The first step of long-term use of what is the extension of the on-site storage of spent fuel in dry flasks unloaded from the reactor storage pool after some years of cooling (21, 22).

From a safety point of view, adequately designed and operated dry storage has the advantage that the probability of a loss-of-cooling accident is extremely small.

Several transport flasks (CASTOR, CONSTAR, TN24) are designed to withstand external events and have been tested for use between on-reactor-site storage and AFRs (Gorleben) (23); they can also be used in Monitored Retrievable Sites (24).

The problems related with heat transfer, criticality and long-term integrity of the flask have already been thoroughly assessed in the framework of direct disposal concepts for spent fuel. The disposal containers are different and are usually designed to withstand very high hydrostatic and lithological pressures, and the materials used in container construction are compatible with the geochemical characteristics of the host rock. Important studies have been conducted in Sweden (25), Germany (26) and the United States (27).

### 2.2.6 Spent fuel reprocessing

All currently existing or planned commercial reprocessing plants use or envisage using the PUREX process (28). This is a solvent extraction process in which the mechanically sheared fuel is dissolved in nitric acid, and counter-currently contacted with tributyl phosphate (TBP) dissolved in kerosene or dodecane. A simplified flow diagram is shown in Figure 2.7. An industrial-size reprocessing plant (4 to 6 THM per day) comprises the following main operations:

- head end process, i.e. receipt of fuel from storage, shearing or decladding, dissolution and fissile material accountability;
- separation of the uranium and plutonium from other actinides and fission products by liquid extraction;
- uranium and plutonium purification;
- uranium and plutonium concentration;
- uranium and plutonium conversion to oxide and intermediate storage;
- fission product concentration and storage;
- treatment of waste streams.

Industrial experience with GCR and LWR fuel reprocessing has been gained in the last 30 years in France, the United Kingdom and Japan (29, 30, 31) since reprocessing is the established fuel management route in those countries.

In France two industrial reprocessing plants (UP1 at Marcoule for GCR fuel and UP2-400 at La Hague for GCR and LWR fuel) have been in operation 30 years and 15 years respectively. Up to now, 4,900 THM GCR fuel and 2,900 THM LWR fuel have been reprocessed. The UP2-400 plant processed both types of fuel fed from two different head ends: the U metal-decladding-dissolution head end and the  $UO_2$ -shearing-dissolution head end (HAO).

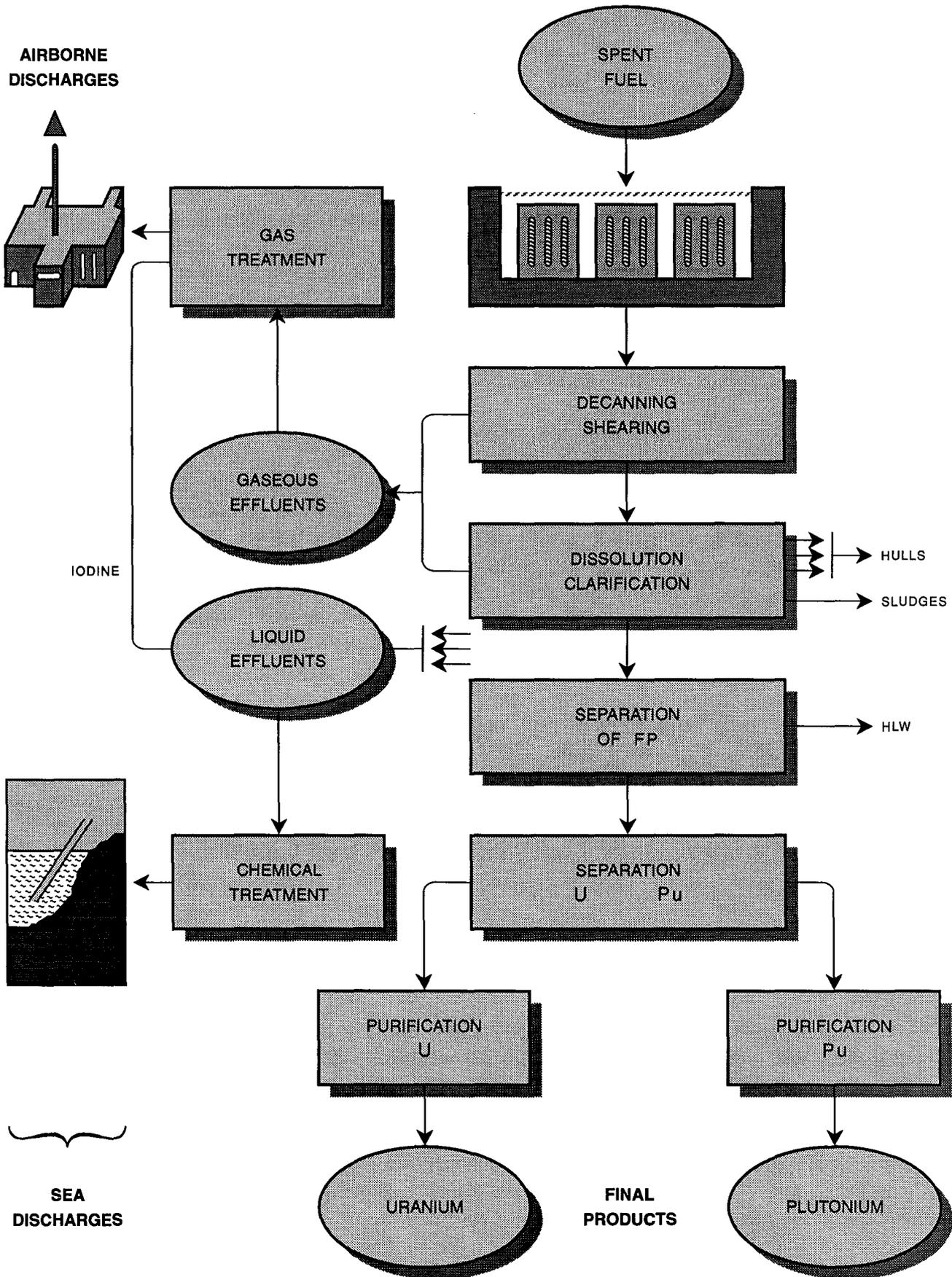
The metal fuel head end of the UP2-400 plant was decommissioned in 1987, but the HAO head end and the extraction plant module with a capacity of 400 THM continue to be operable. A new facility, UP2-800, is being built with a capacity of 800 THM per year. A further LWR fuel reprocessing plant, UP3, with a nominal capacity of 800 THM per year, became operational in 1989 (20). These plants are each equipped with a shearing and dissolution unit and three extraction cycles (uranium-plutonium, plutonium and uranium) based on pulsed-column extraction technology.

A vitrification plant, R7 (part of UP2-800), was put into operation in 1989. A new continuous rotary type dissolver made of zirconium was put into operation in UP3. A second vitrification plant, T7 (part of UP3), began operating in 1992. When all these facilities become operational, the complex of La Hague will have a nominal capacity of 1,600 THM per year.

A pilot reprocessing facility has been constructed at Marcoule. It is used to demonstrate the feasibility of FBR and MOX fuel reprocessing, hulls treatment and gas purification on a 5 THM per year basis (34), and serve for large-scale Research and Development (R&D) purposes.

In the United Kingdom, the reprocessing of spent fuel has been undertaken at Sellafield for more than 30 years with a cumulative throughput of over 30,000 THM which is expected to rise to some 50,000 THM by the end of the Magnox programme.

Figure 2.7 Reprocessing Operations



To serve that programme, a large-scale reprocessing plant was built in the early 1950s. A second reprocessing plant was commissioned in 1964. During the 1970s, this latter plant was refurbished and reinforced in anticipation of the continued operation beyond the turn of this century of the Magnox stations.

Following a public inquiry in 1977, approval was given for the construction at Sellafield of a reprocessing plant, THORP, to deal with oxide fuel. This plant is now at an advanced stage of construction and plans to be in operation by the early 1990s (36). The THORP facility has a capacity of 1,200 THM per year, and an expected lifetime of at least 25 years.

The modern and flexible irradiated MOX fuel reprocessing plant at Dounreay has a throughput of up to 8 THM (U+Pu) per year for fuels of any enrichment or isotopic composition. Eighteen THM have been reprocessed to date, extracting over three tonnes of plutonium. The plant will operate for the foreseeable future reprocessing further fuel arisings from the United Kingdom Fast Reactor. In addition, non-standard foreign LWR fuel is starting to be treated. This facility complements the Dounreay Research Reactor Reprocessing Plant which has reprocessed around 1.7 THM of high-enriched uranium from over 10,000 elements during more than 30 years.

In Japan, the Tokai reprocessing plant with a nominal throughput of 210 THM per year was commissioned in 1977. The Tokai plant had processed about 500 T of spent fuel from different types of reactors by June 1990. In 1985, Japan decided to build an industrial Nuclear Fuel Cycle facility at Rokkasho Mura, in northern Japan. The Tokai and Rokkasho Mura facilities are, together, comprised of an enrichment plant, a low level waste disposal facility (mainly for waste from reactor sites), and a major industrial reprocessing plant of 800 THM per year (including a vitrification plant) and an interim storage facility of waste returned from France and the United Kingdom. The new development in reprocessing was based on French technology with partial inputs from Japanese, German and United Kingdom technology. Japan also initiated a new programme for a fast breeder fuel reprocessing engineering test facility based on up-to-date technology.

In Germany, a pilot plant, WAK, has operated for about 20 years and reprocessed about 200 THM from different types of reactors (37).

An R&D programme has been developed over several years in order to set up new processes for LWR fuel reprocessing and waste treatment (38).

#### 2.2.6.1 *Head end processes*

In a conventional "chop and leach" head end process, mechanical cutting techniques are used to shear the LWR fuel elements into small pieces which fall into a basket located in a dissolver containing nitric acid. The nuclear fuel components (U, Pu, actinides and fission products) are dissolved, but the hulls remain in the basket. The solution, containing finely divided insoluble residues, is transferred to a centrifugal clarifier or filtering device to remove the solids. The clear solution is analysed to determine its nuclear fuel content and treated to adjust the plutonium valency prior to extraction. The leached hulls made of zircaloy or stainless steel are further cleaned to remove the fissile material content as much as possible.

The vapours and gases released from the chopper-dissolver facility are filtered to remove particulates and iodine (especially iodine-129) before they are discharged. R&D work on the removal of krypton-85 has been carried out but has not yet matured to the industrial level. The THORP facility at Sellafield will have the capability of removing carbon-14 as barium carbonate ( $\text{BaCO}_3$ ).

*Safety aspects of the head end operations (40, 41, 42, 60, 61)*

Mechanical cutting techniques have been developed substantially. The fuel assemblies are fed, either vertically as in UP2-400 or horizontally as in UP3, THORP and UP2-800, to the cutting machine which is equipped with either a horizontal or vertical blade. The shear is designed for complete remote handling and maintenance. Redundancy of equipment, specialised maintenance cells and modular design of electromechanical equipment help to reduce outage time and reduce doses to the personnel.

Zircaloy powder is a concern because of its dust explosion danger. Proper design of the cutting head and, where necessary, the provision of an inert gas blanket, together with the fire suppressant effect of the  $\text{UO}_2$  powder and of the large particle size of zircaloy powder, can prevent such an explosion (61).

Dissolution of chopped spent fuel at an adequately slow rate is not a high-risk operation. However, criticality considerations play a major role in the design and layout of the dissolver and in the sizing of the fuel batches. With the use of higher enrichment fuels, criticality safety becomes more important, and the addition of neutron absorbers is sometimes a proper way to ensure safety.

Insoluble residues consist mainly of very finely divided platinum group metals which may contain oxides of plutonium and various co-precipitating fission products. Their thermal output is very high and requires special precautions to be taken during clarification and the subsequent storage of collected residues.

Recent developments in dissolver design and construction have included the use of zirconium as a corrosion resistant structural material; upgraded dissolver construction quality and the introduction of continuous dissolvers (43, 44), e.g. French rotary basket dissolver. Those developments are expected to increase dissolver throughput and reduce the maintenance work. Accountancy associated with continuous operation is watched very carefully. The removal of material deposition and accumulation in vessels and pipes is adequately monitored.

*Head end of MOX and FBR fuel reprocessing plants (45, 46)*

The differences in plutonium content from 0.9 per cent in LWR fuel to nominal 4.5 per cent in LWR MOX fuel, and 15 to 18 per cent in FBR MOX fuel is the most important factor to be considered when designing reprocessing process equipment. Reprocessing of LWR MOX fuel has been carried out on an industrial scale by "diluting" the MOX fuel with conventional LWR  $\text{UO}_2$  fuel. The higher Pu concentration in MOX fuel calls for close attention to criticality controls and process control equipment capable of detecting abnormal dissolutions to avoid oxide accumulations in the plant (47, 48).

Reprocessing of FBR fuel, particularly after a short cooling time, requires specially designed and engineered facilities in order to cope with the very high plutonium content, the high heat output of spent fuel and the potential presence of sodium on and in the fuel pins. Up to now the chop

operation is carried out on single or multiple pins in order to avoid excessive heating of the equipment. Although sodium may be washed off before chopping, the presence of residual sodium may require inerting of the internal chopper space to avoid sparking and possible hydrogen explosions after contact with aqueous solutions or moist atmospheres.

#### 2.2.6.2 Solvent extraction

Currently, large-scale reprocessing plants use an aqueous/organic counter-current solvent extraction process, which is known as the PUREX process (28), or a closely related variant. In general, the first stage separates the uranium and plutonium from the fission products. Subsequent stages separate the uranium from plutonium and purify each of the nuclear products up to the desired specifications. Figure 2.8 shows a general flow sheet of the PUREX process without engineering details.

In the most common version of the PUREX process, the aqueous solution is a nitric acid solution which is counter-currently contacted with 30 per cent TBP (tributylphosphate) diluted with kerosene or dodecane. Uranium and plutonium enter the organic phase, and the fission products and other products remain in the aqueous raffinate. In a second column, the separation of plutonium from uranium is carried out by reduction with excess U(IV) stabilized with hydrazine added to the aqueous stream. It is important to note that U(IV) is prepared from U(VI) by reduction outside the hot facilities. Plutonium (III) is transferred to the aqueous phase to be further purified while the mixture of U(IV) and U(VI) remains in the organic phase. Finally, uranium is stripped from the TBP stream with dilute-nitric acid. These solutions have to be concentrated by evaporation to 300 to 1,000 grams per litre and transferred to the fuel conversion facility.

The plutonium product evaporators have suffered from corrosion due to REDOX processes occurring at the material surface. The welds were corroded and remote repair work has been reported in Germany and other countries. At present the tendency is to switch to more corrosion-resistant materials, e.g. zirconium, tantalum, titanium-5 per cent-tantalum alloy, and chromium-25 nickel-20 steel.

The use of evaporators working under reduced pressure and at a lower boiling point constitutes a very adequate method to increase the equipment lifetime and to improve the overall safety of the distillation operation. The conversion of uranium and plutonium nitrate solution into  $\text{UO}_2$ ,  $\text{PuO}_2$  or  $\text{UO}_2\text{-PuO}_2$  by chemical treatment and calcination has already been briefly mentioned.

Uranyl nitrate solution is concentrated by evaporation and may be calcined to  $\text{UO}_3$ . This product is converted to  $\text{UO}_2$  by reduction in hydrogen. Plutonium nitrate is concentrated by evaporation, and either kept in solution for subsequent transport to conversion facilities (e.g. Dounreay in the United Kingdom) or converted to oxide at the site of the reprocessing plant (France and the United Kingdom).

The conversion of plutonium nitrate to  $\text{PuO}_2$  can be performed according to two chemical processes: oxalate precipitation followed by calcination to  $\text{PuO}_2$ , or peroxide precipitation giving directly  $\text{PuO}_2$ . Figure 2.9 shows the principal steps of both flow sheets.

The oxalate method leads to the production of fine  $\text{PuO}_2$  micro-crystals which can be used as input for the fuel fabrication process. The peroxide method is more suitable as a preparation step for other processes, e.g. Pu metal production.

Figure 2.8 PUREX Flow Sheet

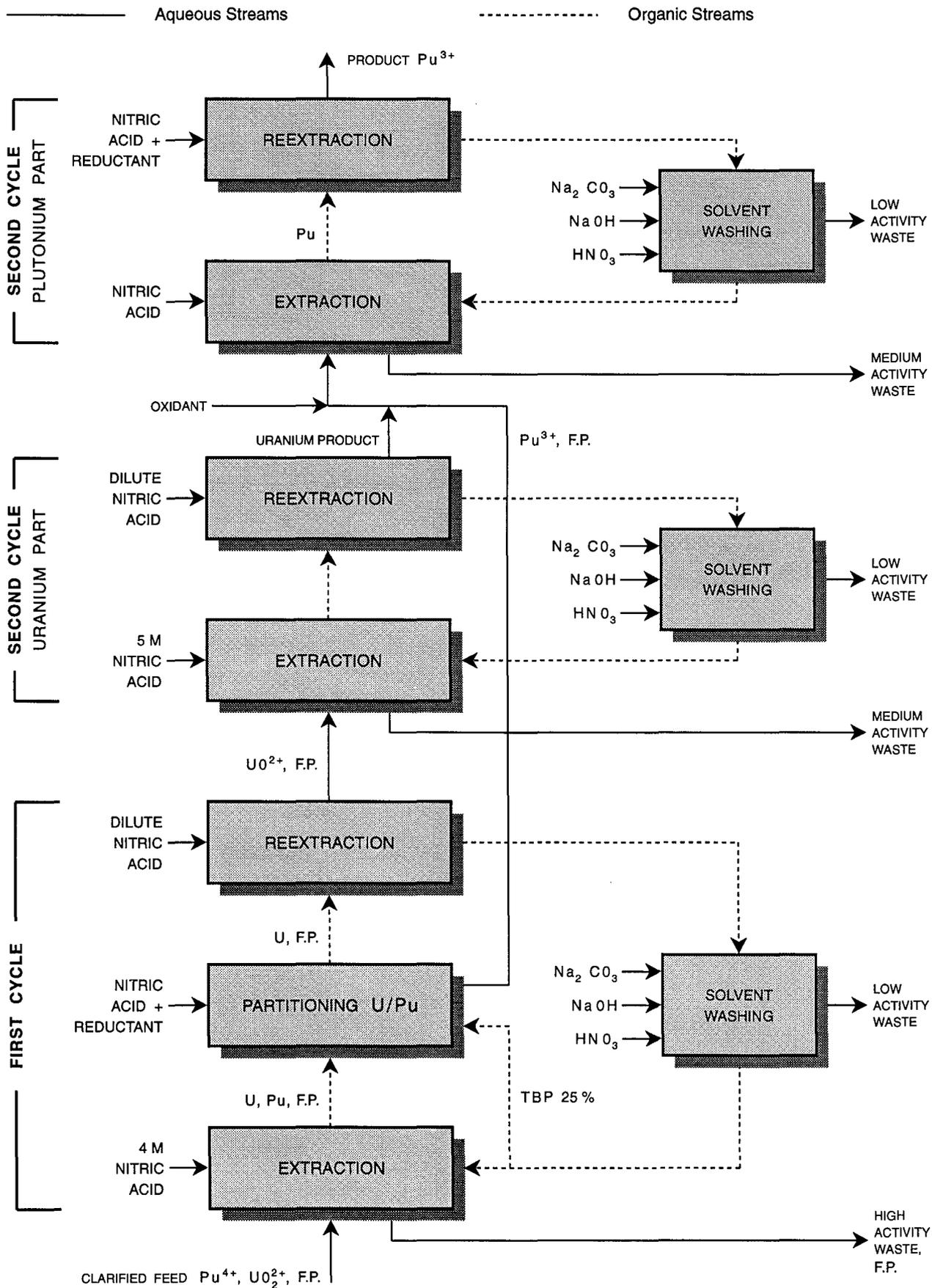
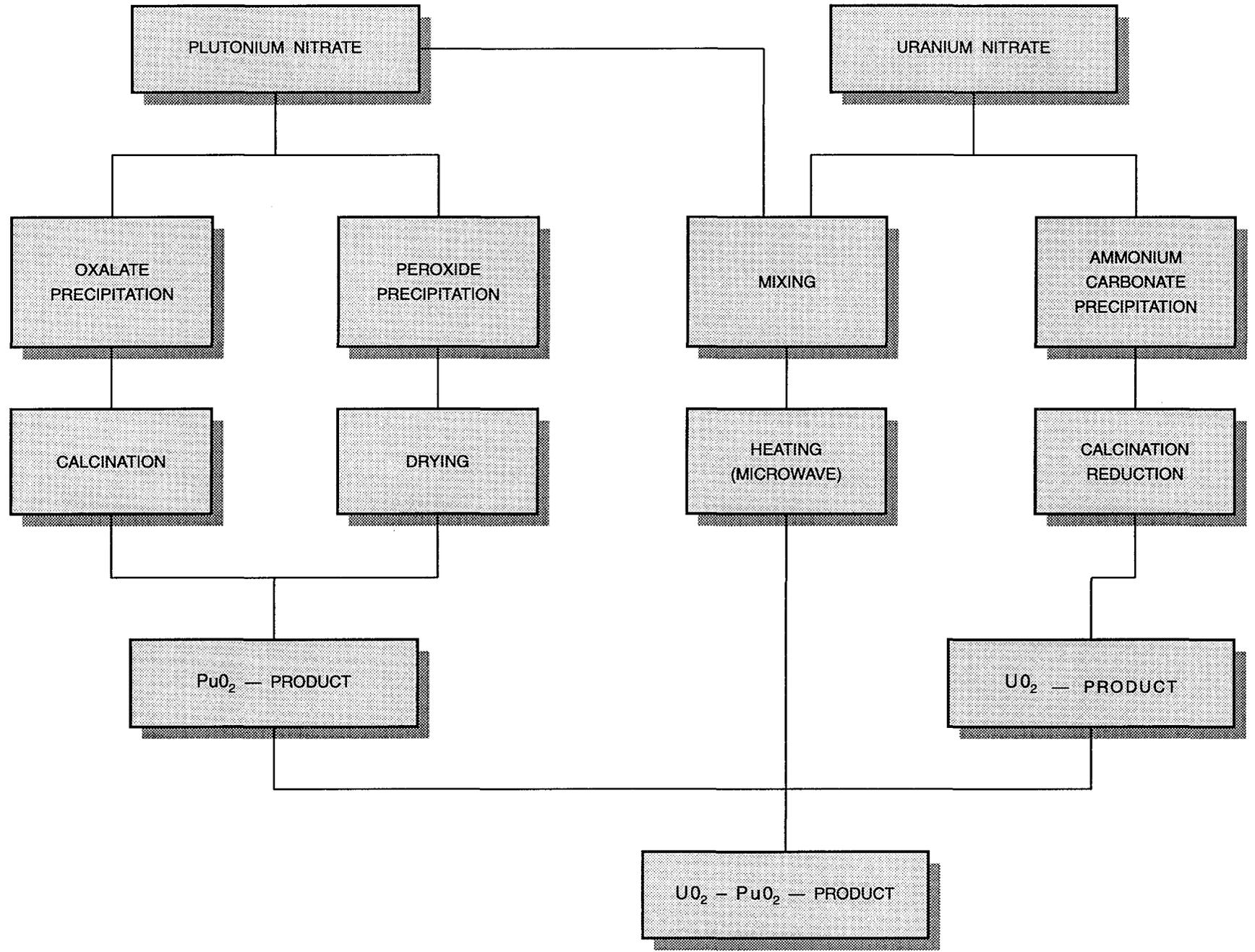


Figure 2.9 Uranium Plutonium Conversion into  $UO_2$ ,  $PuO_2$  and  $UO_2$ - $PuO_2$  for MOX Production



Extensive experience in the production of MOX fuel has been gained in Belgium (Dessel), Germany (Hanau), Japan (Tokai) and in the United Kingdom (Sellafield). In the main, mechanical blending of micronised powders has been employed, however some use has been made of co-precipitation.

The AUPuC conversion method developed in Germany yields directly a U-PuO<sub>2</sub> master blend which is soluble in HNO<sub>3</sub> (17).

Another process developed in Japan starts from uranium-plutonium nitrate solution and calcines the solution by microwave heating to UO<sub>3</sub>-PuO<sub>2</sub>. This product is suitable for intermediate storage or can be calcined under a hydrogen atmosphere to yield U-PuO<sub>2</sub> suitable for fuel fabrication purposes (18).

#### 2.2.6.3 *Plutonium storage*

Plutonium may be stored for various periods of time at different points in the fuel cycle before it is passed to the fuel fabrication plant for incorporation into fresh fuel elements. The stored plutonium may be in various forms:

- a plutonium nitrate buffer store at the product end of the reprocessing plant;
- a plutonium oxide or a master blend of U-PuO<sub>2</sub> at the front end of the fuel fabrication plant or the back end of the reprocessing plant;
- a plutonium fuel element store as MOX fuel.

The storage of plutonium in solution requires special precautions since alpha radiolysis will produce hydrogen which must continuously be removed by ventilation in order to avoid the formation of explosive mixtures. The storage of plutonium solution in any one tank is intrinsically limited for criticality reasons.

Solid plutonium oxide stored for an extended period of time will gradually increase its gamma radiation level due to the decay of plutonium-241 into americium-241. The storage of recycled Pu will have to take into consideration the neutron flux resulting from the (α-n) and spontaneous fission reactions of plutonium-238, plutonium-240 and plutonium-242 (49). The americium-241 content can vary from 1.5 to 3 per cent depending on the plutonium-241 content and the decay time. This important radionuclide inventory emits, except for the regular 5.64 MeV alpha radiation, a wide range of low-energy gamma and X-rays.

The heat emission of 0.55 Watt per gramme plutonium-238 causes increased heat dissipation varying from 2.75 W/kg plutonium (at 0.5 per cent plutonium-238) to 12.6 W/kg plutonium in recycled MOX fuel (at 2.3 per cent plutonium-238).

#### 2.2.6.4 *High level liquid waste (HLLW) concentration and storage*

The high level liquid wastes produced (approximately 5 m<sup>3</sup>/THM) from the first extraction cycles of the PUREX process are normally concentrated by evaporation from 250 to 500 l/THM, depending on the salt content. This HLLW is stored in shielded and cooled tanks for a number of years – the time being dependant on the decision regarding the availability of vitrification facilities. In a typical fuel cycle, the reprocessing operation is scheduled for three to five years after discharge and vitrification follows after an appropriate storage period.

The heat dissipated by the 35 kg of radioactive nuclides per THM amounts to 10 KW/THM one year after discharge from reactor, and it decreases to 1 KW per THM after ten years.

The HLLW tanks are equipped with multiple and redundant cooling coils in order to guarantee that the necessary cooling capacity can be provided at any time without interruption. A single tank may be used for the storage of liquors containing several exa Becquerels ( $10^{18}$  Bq) of fission products and actinides for periods of several decades. Storage of this aqueous solution calls for very reliable and comprehensive safety measures. Radiolysis, sludge precipitation and heat dissipation are the most important phenomena occurring in these tanks. Thus, the reliable safety measures referred to include such features as forced ventilation, sometimes agitation or air sparging, multiple filtration system and adequate monitoring equipment for temperature, liquid level control and leak detection.

The facilities for HLLW storage are protected against external hazards by using conservative design criteria which take into account the most severe recorded hazards (earthquakes, flooding, tornadoes) and by adequate protection systems in case of external fires and aircraft crashes.

### **2.2.7 Radioactive waste management**

After reprocessing, about 99 per cent of the non-volatile radionuclides (fission products, minor actinides) contained in spent fuel are encountered in the:

- high level liquid waste stored in tanks awaiting solidification, e.g. vitrification;
- high level solid wastes e.g. fuel hulls and insoluble residues in storage;
- medium active liquid wastes from the effluent treatment plant, and organic wastes;
- medium active solid waste in storage, e.g. plutonium contaminated solids.

The high and medium active wastes are immobilised for long-term storage and eventual disposal into deep geological repositories.

The supernatant of any Medium Level Liquid Waste (MLLW) flocculation treatment is discharged as low level liquid waste into natural water bodies such as streams or oceans. The sludges are immobilised, e.g. embedded in cement, and stored as solid waste for disposal at ground surface or in deep repositories.

The gaseous radionuclides released during the reprocessing operation – krypton-85, iodine-129, hydrogen-3, carbon-14 and aerosols – are chemically scrubbed and physically filtered by high-efficient particulate air filters (HEPA):

- a fraction of hydrogen-3 and carbon-14 is discharged as liquid effluents;
- iodine (iodine-129) is collected in the scrubbing liquid of the gas purification prior to discharge into the sea or is fixed on silver-impregnated solid sorbent for long-term storage;
- aerosols are trapped by HEPA filters;
- krypton-85 is released at the stack of the reprocessing plants.

R&D work on krypton-85 retention has been carried out in, for example, Belgium and Germany and Germany using cold test units and, in Japan, a hot pilot plant, but they have not as yet reached full industrial scale.

Among the solid high level wastes, the insoluble residues removed from the dissolver solution require particular attention as their activity is very high. These solid materials have to be stored for eventual incorporation into the vitrification process.

Leached zircaloy hulls are generally embedded in cement and stored for some time before disposal in order to reduce the rate of gas formation. Alternative treatment techniques for the hulls, e.g. melting or high-pressure compaction, have been investigated but have not yet been tested on an industrial scale.

Medium level wastes are produced throughout reprocessing from operations such as the washing of solvents, and from distillation operations. These effluents are usually chemically treated by appropriate flocculation techniques in order to remove the bulk of the alpha and beta-gamma emitters. The sludges from such treatment may be stored in tanks, recycled, or embedded in bitumen or concrete. The choice between conditioning media depends on the volume reduction and the final disposal strategy. The main disadvantage of the bitumen process is its potential fire risk.

#### 2.2.7.1 *Solidification of high level waste (HLW)*

Solidification of high level waste through vitrification has reached the industrial stage level. The French AVM and R7 plants (33) and the German PAMELA plants (51) have treated the largest volume of high level wastes, while in Japan and the United States (TVF), new facilities are being constructed or are in the planning stage. A United Kingdom facility (WVP) using the AVM process was commissioned in 1990.

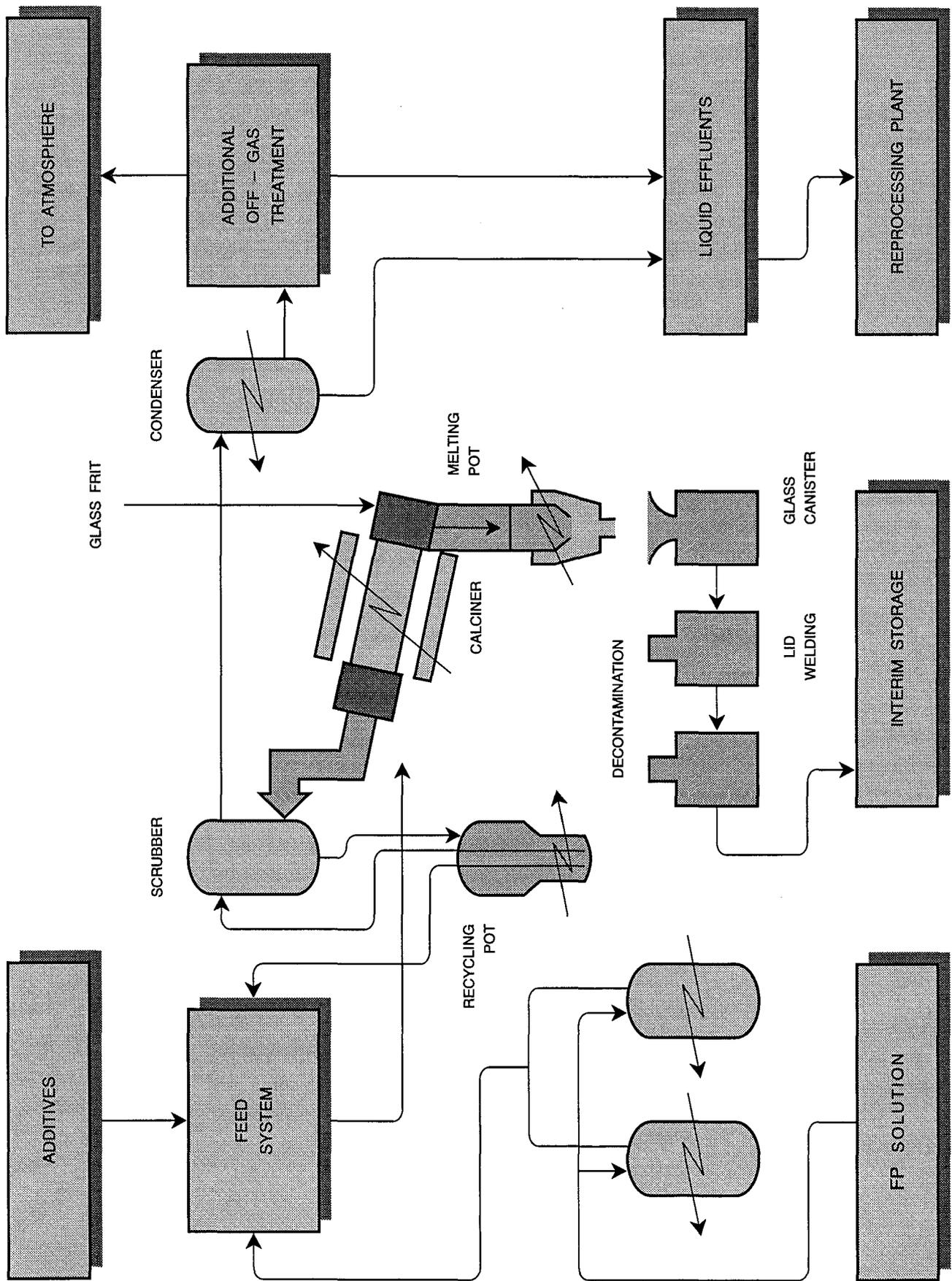
Vitrification transforms HLLW into borosilicate glass by mixing glass forming compounds ( $\text{SiO}_2$ ,  $\text{B}_2\text{O}_3$ ,  $\text{Na}_2\text{O}$ ,  $\text{Al}_2\text{O}_3$ ) with about 11 per cent radioactive oxides and heating the mixture to  $1100^\circ\text{C}$ .

The French Atelier de Vitrification de Marcoule (AVM) process was commissioned in 1978 at Marcoule. It served as a prototype unit for the R7/T7 industrial vitrification facilities constructed at La Hague. The AVM facility has vitrified  $1140 \text{ m}^3$  HLLW with a total activity of  $7 \times 10^{18} \text{ Bq}$  and packaged the wastes into 1450 canisters with a total weight of 500 T borosilicate glass.

In the off-gas system, decontamination factors of  $10^7$  to  $10^8$  have been obtained routinely and the service life of the melting pot was increased from the initial 2,000 hours to almost 6,000 hours. However, repeated repair and maintenance operations on the mechanical equipment carried out over the years caused spread of radioactive contamination throughout the facility. The facility has recently been refurbished.

The new vitrification plant, R7, was commissioned in June 1989 and has produced approximately 1000 glass canisters (each 150 litres) at the end of 1991. R7 and T7 have a nominal capacity of 800 THM per year. Each of these plants features three identical lines, each with a maximum calcination capacity of 60 litres adjusted solution per hour and a vitrification capacity of ten litres glass per hour. The air-cooled storage building can house 4500 canisters at each plant. A schematic diagram of the AVM process is shown on Figure 2.10.

Figure 2.10 The AVM Continuous Vitrification Process



The PAMELA process developed by KfK and built by DWK on the Eurochemic reprocessing site at Mol-Dessel (Belgium) is a liquid-fed ceramic overflow melter heated by Joule effect in the molten glass. The glass formula was adopted to cope with specific HLLW compositions. The glass throughput is 11 kg per hour. This prototype facility was constructed in 1981-84 and since that time has vitrified about 450 m<sup>3</sup> of HLLW producing 940 small canisters of 60 litres and 226 canisters of 150 litres. The off-gas treatment was particularly effective ( $DF = 1 \text{ to } 2 \times 10^8$ ).

The addition of insoluble residues to the HLLW is carefully controlled due to the presence of high concentrations of platinum type metals and molybdates which may locally affect the homogeneity of the glass and create crystallisation centers. Alternatively, the insoluble residues may be stored until their activity level has decreased sufficiently to allow them to be embedded in concrete structures.

PNC's research on vitrification technology began at the Tokai Works in 1975, with small-scale tests. Laboratory-scale hot vitrification started at Tokai's Chemical Processing Facility (CPF) in 1982. Design work on the Tokai Vitrification Facility (TVF) began in 1980 and PNC started cold-test operation in 1992.

An alternative solidification method based on synthetic rock (synroc) formation is under development.

#### 2.2.7.2 HLLW as a resource

High level liquid waste and feed clarification sludge can also be considered as an important resource of radionuclides and radiochemical elements. Partitioning of HLLW into different fractions usable in various technological applications has been thoroughly studied: caesium-137 and strontium-90 can be separated from HLLW and can be used as gamma and heat sources; the platinum group metals (Pd, Rh, Ru) are valuable noble metals useful in the catalyst industry; minor actinides (Np, Am, Cm) constitute the only long-lived alpha-emitting nuclides which can be transmuted partially into shorter lived elements. This approach was studied for several decades in the United States, the United Kingdom, France, Japan, Belgium and at the Joint Research Center (JRC) of the CEC. Recently the partitioning-transmutation option was given new impetus by the setting up of the OMEGA project in Japan (52).

Separation and removal of caesium (as the chloride) and strontium (as the fluoride) from stored defence waste has been accomplished at the Hanford site in the United States. Attempts to use the caesium chloride in double-walled capsules for commercial irradiation have been frustrated by the unfortunate leakage of the caesium from one capsule (63). The other separations discussed above have been studied extensively in France (Castaing Report) (53) and at the JRC (54), but no industrial development has as yet been reported. The overall cost-benefit analysis of the safety and radiological impact of these partitioning-separation processes on the fuel cycle safety are not clear cut and need further analysis.

### 2.2.8 *Transport of radioactive materials (55, 56, 57)*

The transportation of radioactive materials has become a major industrial activity which affects all the nuclear fuel cycle facilities from uranium mining and milling to the final disposal of high level waste in an underground repository. Regulations for the safe transport of radioactive materials are established in the International Atomic Energy Agency (IAEA); each country enforces regulations based on these. In the front end of the fuel cycle, ores and concentrates are transported in large quantities (millions of tonnes per year) on land and sea between the uranium ore producing centres and the uranium refining plants. In the United States and Canada these shipments take place mainly on land while shipping is the most frequent transportation system between Europe, Africa and Australia. Ore is transported on open trucks between the mine site and the milling plant. The ore concentrate (yellow cake) is generally transported by railway or truck in barrels.

Uranyl nitrate hexahydrate (UNH) and  $UF_4$  are solid granular substances which are transported as special nuclear materials by truck or railway at a level of thousands of tonnes per year.  $UF_6$  (at low enrichments) is transported in specially designed and equipped cylinders. Transportation is usually made by railway or by ship.

When needed, enriched  $UO_2$  is transported in specifically designed and fabricated containers (e.g. Bu-J type, used in Japan) between the conversion plant and the fuel fabrication plants by modular loads of 25 kg on large trucks loaded with up to 200 packages per truck.

Fuel elements are dispatched in containers from the fuel fabrication plants to the reactor sites by road, rail, ship or air transport. Additional security precautions are taken when plutonium is transported either as plutonium oxide, mixed oxide or MOX fuel elements. Two fuel assemblies of a FBR can be transported on one truck or railcar. In some countries, e.g. the United Kingdom and Germany, plutonium may be transported as aqueous solution while in other countries, e.g. France, this practice is forbidden. Exceptional precautions are taken when transporting plutonium from one continent to another, e.g. from Europe to Japan, especially during the first major shipment in 1984.

At the back end of the fuel cycle, special transport packages (58), e.g. TN12, TN17, TN24, HZ, EXCELLOX and CASTOR, have been designed and built to transport 5 to 24 spent fuel elements from the LWR site to the away from reactor or the reprocessing plant site. These packages which are massively shielded, provided with neutron shielding and heat dissipation fins, are generally transported by railway and/or ships over large distances and transferred by trucks to the reprocessing plants. For each 1,000 MWe year capacity, about 27 THM of fuel elements have to be dispatched. This amounts to about 9,100 THM per year throughout the world.

Finally, the high and medium level wastes are transported from the reprocessing site to the storage or repository site in specially adapted packages (TN12 or CASTOR type for example) for HLW.

The packages used to ship spent fuel or waste packages have been designed to withstand a train crash. This test has been demonstrated in the United States and the United Kingdom on a real scale and showed that this type of package is fully resistant against such crashes.

Depending on the quantity of fissile materials or radionuclides, the packages are gradually more mechanically resistant and the regulatory precautions more stringent.

### 2.2.9 Decommissioning (59)

All plants, including nuclear reactors and nuclear fuel cycle facilities, have a useful life which is determined by economic, technical and safety factors. These factors contribute to the decision-making process on decommissioning of a facility. In the case of nuclear fuel cycle plants, decommissioning should be considered at the design stage and provisions made to facilitate both decontamination and dismantling.

Decommissioning of facilities occurs in three phases: during Phase I, all radioactive sources and superficial contamination are removed, but all the safety features of the plant remain operational; during Phase II a thorough decontamination of the facility is carried out, all nuclear equipment is removed and occasional surveillance remains to ascertain the continuing integrity of the facility; during Phase III, the facility is completely dismantled or thoroughly transformed for other than nuclear purposes. Radioactive waste resulting from Phase I activities is very similar to that produced by the plant during its active life. Waste produced during Phase II has to be segregated to reduce the volume of nuclear waste which has to be stored for an extended period of time. The segregation is sometimes difficult since some radionuclides (alpha emitters, low-energy beta emitters) are hardly detectable in structural materials. The waste of Phase III is close to the "de minimis" level, where and if applicable, and has to be examined to determine to what extent exemption rules are applicable. Experience has been gained with the dismantling of the PIVER facility at Marcoule in France, the fuel fabrication facility Nukem in Germany, the reprocessing plants Eurochemic at Mol-Dessel in Belgium, and at Dounreay in the United Kingdom.

Research and Development work has been carried out to investigate, among others, the following items (59):

- a) Dismantling of structural components
  - cutting techniques by acetylene burners, plasma jets, arc-melt and arc-saw equipment;
  - thermal scaling of concrete surfaces by plasma torch and microwave.
- b) Decontamination of metal surfaces
  - electrochemical decontamination of metal surfaces;
  - chemical etching of technical components.

The major problem with many of these activities lies in the production of radioactive aerosols and in the volume of secondary waste which is produced and must be treated.

### 2.2.10 Present and proposed fuel cycle facilities

The present status of the fuel cycle facilities in OECD countries and to a certain extent in the world is given in Chapter 9. This review is up to date to 1990 and is based on data from the IAEA, ESARDA and Nuclear Engineering International. Closed-down plants and delayed projects have not been included in the list (1, 2).

In Chapter 2, data concerning mining, enrichment, fuel manufacturing and storage capacities are based on NEA official publications entitled *Nuclear Energy Data* (the "Brown Book") and *Uranium Resources, Supply and Demand* (the "Red Book").

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## **3. Safety Principles**

### **3.1 Safety philosophy**

This section gives an overall picture of the safety philosophy which is adopted to ensure high standards of safety in the nuclear fuel cycle. All nuclear fuel cycle facilities are subject to regulatory control which requires installations to be designed, constructed, commissioned, operated and decommissioned with due regard to the safety of workers and to the general public.

The basic safety principles relating to radiation protection are based on international understanding of the effects of ionizing radiation. Most countries have adopted the recommendations of the International Commission on Radiological Protection (ICRP) (1, 2) which advocates three basic principles:

- No practice shall be adopted unless its introduction produces a positive net benefit.
- All exposures shall be kept as low as reasonably achievable (ALARA), economic and social factors being taken into account.
- The dose equivalent to individuals shall not exceed the limits established by the appropriate regulatory or licensing body in that country.

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

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

The remaining sections of this chapter describe the regulatory framework which enables plants to be controlled, the treatment of radiological safety and plant malfunctions, the approach used to assess the behaviour of plants under accident conditions and the reporting of nuclear incidents.

### **3.2 Regulatory framework**

In this section the role of the regulator and the effects of the regulatory framework on the safety of nuclear fuel cycle facilities are addressed. There are differences in regulatory practices in each Member country and therefore only typical examples of these differences are described. The

roles of governments and the regulatory bodies are discussed together with the steps taken through the regulatory process to ensure that operators of nuclear facilities pay due attention to safety.

### **3.2.1 *Governmental responsibilities***

Nuclear safety policy in addition to being set and enforced by the regulatory bodies of the respective governments, is also a matter of international concern. A recent survey (21) of countries engaged in nuclear power programmes has shown that, in most countries, consistent policies relating to nuclear safety regulations have been developed. A more comprehensive description of the role of governments in the regulation of nuclear industries is given in the IAEA Safety Series Code N°. 50-C-G (Rev. 1) (3) which, while primarily aimed at nuclear power plants, is recognised as being applicable to other fuel cycle activities.

It is important to recognise that the responsibility for the regulation of nuclear fuel cycle facilities rests with Government. However, the responsibility for the safety of workers in these plants and for the general public rests with the operator of the facility. This shared responsibility provides the basis for the effective regulation of the nuclear industry so that society can enjoy the benefits of nuclear power without being exposed to intolerable risks (4). To tolerate a risk does not mean it is negligible, nor can it be ignored. Any possible risks to plant workers and the general public must be kept at levels in accordance with national regulations and the ALARA Principle.

Governments define and implement the legal framework within which the operator of the facility works. The objectives of such legislation are generally:

- to provide the statutory basis for establishing a regulatory body to oversee the safety of the industry;
- to provide the regulatory body with powers to establish and enforce the necessary regulations with respect to safety;
- to provide the legal basis for ensuring that nuclear fuel cycle facilities are sited, designed, constructed, commissioned, operated and decommissioned without undue radiological risk to site personnel or the general public and that proper regard is paid to the protection of the environment;
- to ensure the provision of adequate financial indemnification for third parties in the event of any harm that might arise as a result of operations.

### **3.2.2 *Responsibilities of regulatory bodies***

A regulatory body within each respective country has responsibility for full governmental surveillance and control of all matters relevant to the safety of nuclear installations. These responsibilities in general include the siting, design, construction, commissioning, operation and eventual decommissioning phases of the life of an installation. Governments, in general, ensure that the regulatory body is as independent as practicable of any other government agency. This is to ensure that the regulatory body is not faced with the possibility of conflicting requirements which could compromise its position in the upholding of safety standards. The regulatory body is, in the majority of countries, also independent of the operators of the fuel cycle facilities, the vendors of nuclear equipment or other related organisations.

The regulatory body has the responsibility to regulate the operations of the nuclear fuel cycle facilities and it does this through its statutory authority given to it by Government. The power to

regulate the industry is exercised via a licensing process which in general ensures that no person can site, design, construct, commission, operate or decommission a nuclear facility without a licence. The power of the regulatory body to grant licences carries with it certain responsibilities which ensure that licences are granted and enforced in accordance with well-defined safety principles and associated criteria. The safety principles and criteria which are used to assist the decision-making process in the regulatory body relate to such topics as:

- radiation protection;
- environmental protection;
- nuclear safety;
- physical protection;
- structural integrity;
- internal hazards such as fire and explosion;
- external hazards, both man-made and natural;
- nuclear material control and accounting;
- handling and transport of nuclear materials;
- personnel qualification and training;
- inspection, testing and maintenance;
- plant modifications;
- quality assurance;
- emergency planning.

The regulatory body has a further responsibility to ensure that it has the necessary structure and expertise to deal effectively with its main functions such as review and assessment, licensing, inspection, enforcement, development of regulations and guides, and legal and administrative support.

### **3.2.3 *Regulations and guides***

The regulatory body makes use of regulations and guides as an effective method to obtain the necessary high degree of safety required in nuclear installations. Regulations give a clear prescriptive statement of what is required of the operator but they can also be inflexible and can often take long periods of time to change. It is because of this that regulations have limited application in the day-to-day regulation of nuclear fuel cycle plants where flexibility is required to accommodate changes in processes and operations. This flexibility is provided by the use of the licensing process. Guides, however, do not suffer the same problems as regulations as they are usually non-mandatory documents which are intended to amplify and clarify safety principles or other matters relevant to safety.

### **3.2.4 *Licensing***

Licensing processes have been developed to give a flexible and yet, at the same time, firm control over nuclear safety related activities. A licence is a legal document which:

- authorises a specific activity or facility;
- establishes requirements and conditions governing the performance of the activity; and
- where appropriate, places time limits on the validity of the authorisation.

The licensing process can be considered as an on-going process which is carried out throughout all phases of the facility life. Before a licence is granted, the applicant is required to

submit a detailed report on the safety analysis of the proposed plant. This submission is reviewed by the regulatory body to see if an adequate case has been made for the siting of the plant and for the identification and assessment of the risks it poses to both site personnel and the general public. When the regulatory body is satisfied that an adequate safety analysis and evaluation has been made, a licence can be granted.

At the time the licence is granted it is possible that the detailed design work may not be complete and hence detailed design may proceed in parallel with construction. The regulatory body can control this situation by placing "hold points" in the construction programme or by issuing short-term or "limited" licenses to ensure that design work related to safety systems is done on time. When construction is completed, the operator of the plant is often required to apply for a further authorisation to commence commissioning. At this stage the regulatory body will require a report showing that the plant has been constructed in accordance with its specifications and that it is acceptable to commence commissioning. This report will also identify the commissioning requirements in order to demonstrate the operability and safety features of the plant.

When the plant has been successfully commissioned, the operator is then required to seek an authorisation to commence routine operations. This application will often be accompanied by a pre-operational safety report on the results of the commissioning tests and the plant's operating procedures. When the regulatory body is satisfied with the case it will authorise routine operation.

Operating characteristics of a facility are controlled by the conditions of its licence. The procedures for maintenance, inspection and testing are to be used by the operator of the facility to ensure that safety is maintained. The licensing process also ensures that the safety margin for the plant is periodically reviewed to take account of any changes in the facility and possible evolutions in the safety standards.

### **3.2.5 *Review and assessment***

One of the major features of the licensing process is the requirement for the operator, i.e. the licensee, to produce documentation to demonstrate the safety of his operations. Hence the licensee submits safety documentation to the regulatory body at various stages during the life of the plant. The regulatory body determines the adequacy of the safety proposals.

The IAEA Code N°. 50-C-G (Rev. 1) (3) gives a comprehensive description of review and assessment requirements. The regulatory body not only reviews and assesses the technical details and safety justification for a given plant, it also looks at the licensee's overall management of safety issues. Review and assessment is, therefore, an important and on-going part of the licensing process and, as such, is a fundamental requirement in nuclear safety policy.

### **3.2.6 *Regulatory inspection and enforcement***

Another fundamental requirement of the regulatory framework used to ensure adequate nuclear safety standards is the role of the regulatory body in inspection and enforcement. The regulatory body is responsible for checking that the licensee is fulfilling the conditions set out in his licence. This is done by regulatory inspection.

Inspectors have powers of entry to licensed sites and powers to obtain any relevant information relating to the siting, design, construction, commissioning, operation and decommissioning of the plant. Regulatory inspections can be either announced or unannounced depending upon their purpose and objective. The principal objectives of inspection are to check compliance with licence conditions and to monitor the licensee's safety performance.

The method of inspection is a matter for individual countries as geographical location and the number of installations to be inspected can influence the inspection philosophy. Some countries have inspectors based at the regulatory body headquarters who carry out periodic visits to the nuclear site. Other countries have opted for the inspector being based at the site and rely on a reporting back procedure. Where there are logistic difficulties, some other countries have adopted the concept of regional offices to enable the inspector to be sufficiently close to their sites to prevent the need for permanent location at the site.

The legal power to enforce compliance comes from the statutes used to establish the regulatory body. The methods of enforcement vary depending upon the nature of the violation. Minor non-conformances can be handled by a written warning from the regulatory body. More serious problems may require the regulatory body to use its powers to order the licensee to curtail or modify its activities. In the event of chronic or serious non-conformance, such as significant harm to the workforce or the general public arising from the licensee's negligence, the regulatory body could revoke the licence or a particular authorisation.

Enforcement actions can result in penalties of one form or another. The revocation of an authorisation or a licence, or a direction to curtail a particular activity, would almost certainly mean an economic penalty for the licensee. The regulatory body can, in some cases, have the authority to impose direct economic penalties via fines. Also, in some countries, the regulatory body can institute prosecution of the licensee in the Courts for serious offences.

### **3.2.7 *Quality assurance***

The IAEA Code of Good Practice N°. 50-C-QA (1979) gives a comprehensive description of quality assurance criteria for the safety of nuclear power plants. Most of these criteria may also apply to nuclear fuel cycle facilities. The regulatory body has the responsibility to regulate quality assurance related to safety for siting, design, construction, commissioning, operation and eventual decommissioning of nuclear fuel cycle installations. It ensures that quality assurance systems and programmes are established by the operators, their contractors and suppliers and are enforced in accordance with safety regulations.

### **3.2.8 *Emergency preparedness***

The purpose of establishing a regulatory framework to oversee the safety of nuclear fuel cycle facilities is to provide the public with the assurance that such facilities will be designed, constructed, commissioned, operated and decommissioned with sufficient safety margins such that the risks to site personnel, the public and the environment are acceptably low. Nevertheless, in spite of all the precautions taken, there is the remote possibility of failures or accidents which could produce an emergency situation. It is therefore a fundamental requirement that adequate preparations are made, and maintained, at local, national and international levels to deal with such situations.

Emergency preparedness involves the participation of many government agencies including the regulatory body. The regulatory body is often responsible, in conjunction with other government bodies, for ensuring that an adequate emergency plan exists and emergency preparedness is maintained. The licensing process gives the regulatory body the powers to ensure that the licensee's emergency plans are regularly exercised. Failure to demonstrate adequate emergency preparedness would constitute a serious licence non-conformance.

### **3.3 Radiological safety**

This section examines the safety assessment procedures carried out to ensure a high standard of safety against the potential hazards of radioactive contamination by the nuclear fuel cycle. All fuel cycle installations are subject to safety regulations specific to each nation, and are designed in compliance with these regulations, taking into account internal hazards such as fire, explosion, radioactive contamination and criticality, and external hazards such as seismic events, extreme weather conditions, flooding, aircraft crash, fire, explosion and missiles.

In most of the nuclear fuel cycle facilities, such as uranium mining, milling, conversion, enrichment and fuel fabrication, the radioactive materials to be handled are dispersed throughout the facility and, therefore, less concentrated compared with those in the nuclear reactors. Furthermore the nuclear fuel cycle facilities, including reprocessing facilities, are operated in rather low temperature and pressure conditions and in subcritical conditions so that the potential energy is not sufficient to permit abnormal conditions to rapidly develop into dangerous situations.

#### **3.3.1 Radioactive contamination**

Protection against radioactive contamination takes the following considerations into account:

- the physical form of the material (massive solid, powder, solution, gas);
- the chemical property of the material (stability, reactivity, volatility);
- the inventory and composition of the material;
- the specific activity of the material and half-life of the radionuclides that it contains as well as the nature and energy of the radiation which it emits; and
- the behavior of the material within the body (biological half-life, concentration in specific organs).

Determination of the inventory of the process materials is an essential first step in any safety assessment.

The control of the spread of radioactive materials is maintained by containment, using physical barriers, gas flow control and clean up systems. Containment may consist of:

- the process equipment itself;
- a surrounding cell or glove box;
- the room or building in which the plant itself is located.

Ventilation control of the containment with the discharge of exhaust gases through a stack after a gas cleaning process, e.g. scrubbers, chemical traps, high-efficiency filters or electrostatic precipitators, can reduce environmental discharges of radioactive material to low levels.

The buildings in which radioactive materials are processed are also divided into zones of increasing degrees of potential radioactive contamination. The ventilation systems are arranged so that the flow of air is from lower to higher regions of potential contamination, thus helping to contain the active material. Access of personnel to each of these zones is controlled and appropriate protective clothing specified. The zones are monitored for surface and air contamination at a frequency appropriate to ensure safety.

### **3.3.2 *Radiation exposure***

Personnel in all sections of the fuel cycle are protected against external radiation (i.e. against radiation sources outside the body).

There are three means of maintaining external radiation doses at acceptable levels:

- by shielding;
- by controlling the distance between personnel and the source of radiation;
- by controlling the exposure time of personnel.

Handling of recycled fuel may pose other radiation control problems due to a change in isotopic composition of uranium, through natural decay and to the presence of plutonium and of some gamma emitting impurities.

In fuel cycle plants, shielding is designed to reduce dose rates to acceptable levels in regions normally accessible to personnel, and permanently installed interlocks and radiation monitors are used to prevent accidental withdrawal of radioactive material from behind the shielding.

### **3.3.3 *Environmental impact and pathways***

When considering the impact of fuel cycle processes on the environment, we are concerned primarily with the pathways for, and mechanisms of, release of dispersible radioactive material. The pathways are those by direct radiation from the cloud, or by transfer from contaminated ground, or by skin penetration, by inhalation, by ingestion or absorption, and through the food chain.

The movement of radioactive material between fuel cycle sites utilises public highways, railways and sea routes, and during transport the material may pass through significant centres of population. Potential accidental release of radioactive material during transport therefore requires special consideration. In the case of ocean transport, sinking accidents are taken into consideration, and in the case of air transport the consequences of aircraft crash are assessed.

## **3.4 *Abnormal occurrences***

This section briefly refers to non-routine operational situations, such as leaks, instrument malfunctions, and loss of electric power.

Potential mechanisms for producing abnormal situations can be classified as:

- internal events such as physicochemical, nuclear, mechanical, instrumentation and control systems, operator error, loss of electrical power;
- external events such as floods, earthquakes, aircraft crashes.

### **3.4.1 Physicochemical mechanisms**

Many sections of the fuel cycle involve energy releasing or absorbing processes, e.g. dissolution of uranium in nitric acid, evaporation. Many vessels have high-heat ratings resulting from fission product heating, and require cooling. Hydrogen and oxygen are also produced when radiolysis occurs.

#### *Fire*

Flammable material and oxygen are present at certain points in the fuel cycle although often a considerable amount of additional energy is necessary to start combustion. However, since fire is a critical mechanism for the dispersion and release of radioactive materials, due consideration is given to its prevention, detection, minimisation of its escalation, and to its extinction. Various plant ancillary systems, including those concerned with off-gas clean-up, are also protected against fire. When deciding on fire-fighting or fire prevention systems, thought is given to possible adverse consequences arising from their use, e.g. the safety of a complex plant associated with inert atmosphere systems, injection fire-fighting systems which can cause pressurisation of a cell or a glove box, production of toxic materials, dispersion of radioactive materials within the containment, and moderation of fissile materials which could lead to a criticality accident and subsequent clean-up difficulties.

#### *Explosion*

In some areas of the fuel cycle, the potential for the creation of an explosive atmosphere exists through the production of hydrogen by radiolysis, and by chemical or corrosion reactions through the presence of finely divided, oxidizable dusts or pyrophoric materials; and through certain nitric acid/heavy metal/organic-matter reactions. An explosion may cause either physical damage such as breach of containment or system pressurisation resulting in an unplanned discharge of radioactive material. In such situations, consideration is given to the use of inert gas atmosphere or dilution systems, and to the ability of the containment systems to withstand the explosions which may occur.

#### *Equipment failure*

In any facility there is the possibility of equipment and plant failure, resulting in the interruption of some of the essential safety functions, such as containment or cooling. Furthermore, unexpected chemical reactions may occur, for example when normally separated sections of a plant are accidentally interconnected by siphon or a leak path.

To guard against the adverse effects of such failures, various design features are incorporated such as redundant reliable systems, enhanced monitoring equipment and strict quality assurance.

### **3.4.2 Criticality**

A criticality accident is an uncontrolled nuclear chain reaction occurring in an assembly or facility containing fissile materials, and occurs when the geometry or composition of such an assembly is changed such that the rate of neutron loss due to capture or leakage is in balance with the neutron production rate from fission. Its consequences are a sudden increase of the neutron flux in the system and a corresponding energy release, generally rather low, but with the formation and possible dispersion of fission products, and the emission of an intense gamma and neutron flux. The most important consequence of such accidents, in an unshielded situation, is the very high direct radiation doses (usually fatal) which might be sustained by operating personnel in the immediate vicinity, if this personnel was neither warned nor properly protected.

Criticality, criticality control and criticality incident experiences are discussed in Chapter 4.

### **3.4.3 Mechanical failures**

Mechanical failures leading to a potential loss of containment or cooling may be gradual, such as due to corrosion or erosion, or sudden, e.g. when a heavy load drops on a valve, collision of a crane with a pipe bridge, or the damage to a cell window during a manipulator operation. The gradual failures are detected and repaired before serious consequences arise while the sudden ones will require immediate response from the operators and protective systems.

### **3.4.4 Malfunction of instrumentation and control systems**

Computer control systems have been introduced in nuclear fuel cycle facilities and are frequently used to reduce human errors. But such systems also have the possibility for malfunctions. Some malfunctions are attributed to mechanical system failures, while other malfunctions can be attributed to software systems and the computer itself.

Malfunctions in mechanical systems are caused by breakdowns in the mechanical systems themselves, mainly caused by failures in parts such as sensors, connectors or actuators. The operation of these mechanical parts is periodically checked and maintenance usually prevents failure in these parts. Malfunctions caused by fundamental operating software are usually prevented by pre-operational and validation tests. An important part of computer control system design consists of redundant systems to improve their reliability, thus minimising downtime or operational loss of control caused by single failure in the computer system. Furthermore, mechanical systems and their associated control systems are, as far as possible, of safe design (i.e. that a failure leads to a safe state).

### **3.4.5 Operator error**

The degree of automation or computer control in the fuel cycle plants is very advanced, but there is ongoing intervention by man, and thus the possibility for operator error still exists. It is therefore possible that an operator error, or inaction, can either generate or cause a deterioration in maintaining system safety or a failure to take corrective actions. Plant and processes are designed and the plant operated in order to minimise this possibility. Operators are furthermore adequately trained to enable them both to carry out routine operations and to respond to abnormal conditions.

### **3.4.6 External events**

When reviewing the safety of nuclear fuel cycle facilities, the role of external events as possible causes of accidents is examined. External events may be natural, e.g. floods, earthquakes, forest fires, tornadoes, or man-made, e.g. aircraft crashes, missiles, unconfined vapour cloud explosions from adjacent sites or transport routes. Site selection and plant design play an important part in minimising these risks, and certain controls on off-site activities, such as overflying the site, may be required. Although an aircraft crash on the relatively small "target" of a nuclear fuel cycle site would have a very low probability, protection against it may have to be provided depending on site conditions and facility operations. Decisions regarding the design measures which are required to minimise possible external event induced hazards are based to a large extent on statistical and historical data.

### **3.4.7 Loss of electrical power**

In order to ensure that safety features can fulfill their function, the plant's electrical power supply is designed so that the necessary power can be supplied by either off-site or on-site power sources in the event of an emergency. In particular, special provisions are taken to ensure sufficient redundancy and diversity for both off-site and on-site sources.

## **3.5 Safety assessment**

Radiation doses to the public from the routine authorised release of radioactive effluents from nuclear fuel cycle plants are assessed as completely and as realistically as possible, and operating limits are imposed on the facilities in the licences. In section 7.3, an overview is presented of the safety records of fuel cycle facilities and their compliance with regulations.

In the assessment of potential internal and external abnormal events, the deterministic technique has generally been used. In using the deterministic assessment approach, a "design basis event" is selected and evaluated to prove the adequacy of the facility design. A design basis event is an artificially supposed event, considered as the upper bounding event, against which measures are taken when designing the facility. As a result of its evaluation, a number of other events or incidents of lesser consequence will be mitigated by the safety measures. Another approach in assessment, the probabilistic technique, has been used supplementarily. In the selection of external events, statistical and historical data are considered.

The following are considered in the safety assessment:

- safety-related experience from plant operation including that from relevant conventional industries;
- identification of potential abnormal events;
- estimation of the possibility of the events;
- determination of scenarios and assessment of the consequences of such events in appropriate units.

### **3.5.1 Identification of fault situations**

The safety assessment requires the application of professional training, experience and judgement using techniques such as HAZOPS (5), failure mode and effect analysis (6, 7) or cause/consequence diagrams, to ensure thorough examination of the whole system. The aim is to identify all mechanisms which may create potentially hazardous situations. These may differ in nature – physical, mechanical or chemical properties – or may be due to operator error and therefore may cover a range of occurrences such as over-pressure, loss of vacuum or inadvertent mixing of reagents. In the deterministic approach, possible design basis events and their scenarios are examined according to the above considerations.

### **3.5.2 Fault tree analysis**

Estimation of the possibility of a design basis event relies on experience and engineering judgement. In some cases, a probabilistic technique using fault tree analysis might be used to determine its probability quantitatively.

Fault tree analyses are detailed deductive analyses of specific situations which identify conditions that could lead to a stated event, usually a failure. After the undesired events (TOP EVENTS) are identified, the analyst must construct a fault tree that models the system conditions that can cause the undesired event. In this case, care should be taken to consider human errors and common failure modes. The fault tree graphically traces the sequences of individual failures to the final event (8).

The purpose of constructing a fault tree is to determine the probability of the top event. This is done by the application of available fault rate data and statistical techniques to the sub-system. Data on operator response is frequently a source of much debate in this context.

### **3.5.3 Assessment of consequences**

A design basis event is determined according to the scenarios covering the effects of significant possible events. In assessing the consequences, technically approved analytical models are used together with reliable data. The behaviour of process systems during the event is determined in order to fully evaluate the possible consequences. It might be unrealistic to assume that, for example, in the event of a fire, the total plant inventory of radioactive materials would become airborne, but a knowledge of the fraction which could be released would be important. Thus such a situation would require selecting specific values for:

- a) the fraction of the total inventory of the affected unit which will be involved in the incident;
- b) the fraction of a) that will be released from the bulk of the material;
- c) the fraction of b) that will escape from the containment to the working areas or to the outside environment.

In some situations, accident progression occurs over a long period of time, thus allowing remedial actions to be taken. A typical example of this is the loss of cooling in an active liquid waste storage system. In this case, it would be necessary to evaluate the system's temperature and

the activity release transients. These have to be related to the possibility of cooling restoration or other remedial actions which might reduce or prevent releases.

Some information is available (9, 10, 11, 12, 13, 14, 15, 16) on these topics and where relevant this can be applied to specific accident assessments. In some cases, application of small-scale experimental data to plant conditions may be valid. Information on actual plant incidents is frequently not well collated and related to design parameters.

It is also necessary to calculate the dispersion through the biosphere of any radioactivity which is released from a plant containment and the radiological consequences for the surrounding population, generally using appropriate computer programmes (17, 18).

### 3.6 Event severity scales

The International Nuclear Event Scale (Figure 3.1) is being established in IAEA and OECD/NEA Member countries by reflecting experiences gained from the use of similar scales in France and Japan and from considerations of possible scales in several other countries. The Japanese event severity scale was replaced by the International Nuclear Event Scale on 1 August 1992.

The international scale is divided into seven levels. The lower levels are termed incidents (1 to 3) and the upper levels (4 to 7) accidents. Events which have no safety significance are classified as below scale/level zero. Industrial accidents or other events which are not related to nuclear plant accidents are not classified on the scale; these are termed "Out-of-Scale" (19). Events are considered in terms of three broad categories; off-site impact, on-site impact, and safety system degradation.

- The first category applies to events resulting in off-site release of radioactive material. Understandably, the public is most concerned with such releases. Level 7, the highest on the scale, corresponds to a major nuclear accident with widespread health and environmental consequences. Level 3, the lowest rating in this context, represents a very small release that would result in a radiation dose to the most exposed members of the public equivalent to a fraction of the prescribed annual dose limit for the public. Such a dose is typically about a tenth of the average annual dose from exposure to natural background radiation.
- The second category considers the on-site impact of an event. The range is from level 5, typically representing a situation of severe damage to the facility, down to level 3 at which there is major contamination and/or overexposure of workers.
- The third category applies to events involving the degradation of a plant's safety system. All plants are designed such that a succession of safety systems acts to prevent major on-site and off-site impacts. The defence-in-depth considerations classify events as levels 3 through 1.

An event which has characteristics represented by more than one criterion is always classified at the highest level according to any one criterion.

The scale has been designed for use in nuclear power plants, but its application is also recommended to classify events at nuclear fuel cycle facilities, giving a means for promptly communicating and facilitating common understandings from the nuclear community to the media and the public.

The French severity scales are established for both power reactors and the fuel cycle installations. Events classified on these scales relate only to nuclear or radiological safety. The French scale for fuel cycle facilities (Figure 3.2) is graded from 1 to 6, level 6 for the most serious accidents, and level 1 for the least important incidents. The three accident levels are differentiated by the magnitude of radiation risk affecting the public, and the three incident levels are characterised by the amount of radioactive material released from stacks which have an allocation within the authorised annual limit. Functional troubles are also taken into consideration when assigning the incident level (20).

Figure 3.1 : The international nuclear event scale for prompt communication of safety significance

LEVEL	DESCRIPTOR	CRITERIA	EXAMPLES
5	MAJOR ACCIDENT WITH OFF-SITE RISK	<ul style="list-style-type: none"> <li>External release of radioactivity resulting in a dose to the most exposed individual off-site of the order of millisieverts. With such a release, off-site protective measures would be generally required, possibly for long time periods.</li> <li>Significant damage to the nuclear facility. Such an accident might include damage to nuclear core leading to major on-site releases, and/or a partial core melt in a power reactor and consequent release of fission products.</li> <li>Incidents of one or more systems which result in an event which offers a high probability of core damage.</li> </ul>	<p>Three Mile Island, PA, USA, 1979</p> <p>Chernobyl NPP, Ukraine, 1986</p> <p>Windscale Reprocessing Plant, UK, 1957</p> <p>Sanjona NPP, France, 1980</p> <p>Granville NPP, Canada, 1982</p> <p>Atucha NPP, Argentina, 1982</p>
4	ACCIDENT WITHOUT SIGNIFICANT OFF-SITE RISK	<ul style="list-style-type: none"> <li>External release of radioactivity resulting in a dose to the most exposed individual off-site of less than a few millisieverts. With such a release, the need for off-site protective measures would be generally not required, possibly for short time periods.</li> <li>Significant damage to the nuclear facility. Such an accident might include damage to nuclear core leading to major on-site releases, and/or a partial core melt in a power reactor and consequent release of fission products.</li> <li>Incidents of one or more systems which result in an event which offers a high probability of core damage.</li> </ul>	<p>Windscale Reprocessing Plant, UK, 1957</p> <p>Sanjona NPP, France, 1980</p> <p>Granville NPP, Canada, 1982</p> <p>Atucha NPP, Argentina, 1982</p>
INCIDENTS 3	SERIOUS INCIDENT	<ul style="list-style-type: none"> <li>External release of radioactivity above authorised limits, resulting in a dose to the most exposed individual off-site of the order of tenths of millisieverts.* With such a release, off-site protective measures may not be needed.</li> <li>On-site events resulting in doses to workers sufficient to cause acute health effects and/or an event resulting in a severe spread of contamination for example a few thousand terabecquerels of activity released in a secondary containment where the material can be returned to a satisfactory storage area.</li> <li>Incidents in which a further failure of safety systems could lead to accident conditions, or a situation in which safety systems would be unable to prevent an accident if certain initiators were to occur.</li> </ul>	<p>Vandellios NPP, Spain, 1989</p>
2	INCIDENT	<ul style="list-style-type: none"> <li>Incidents with significant failure in safety provisions but with sufficient defence in depth remaining to cope with abnormal situations.</li> <li>An event resulting in a dose to a worker exceeding a statutory annual dose limit which is not due to the presence of significant quantities of radioactivity in the workplace in areas not protected by design and which requires corrective action.</li> </ul>	
1	ANOMALY	<ul style="list-style-type: none"> <li>Incidents involving the operation of safety systems, which are capable of preventing a release of radioactivity above authorised limits, but which are not fully effective in preventing such a release.</li> <li>Incidents involving the operation of safety systems, which are capable of preventing a release of radioactivity above authorised limits, but which are not fully effective in preventing such a release.</li> </ul>	
BELOW SCALE/ZERO	DEVIATION	NO SAFETY SIGNIFICANCE	

\* The doses are expressed in terms of effective dose equivalent (whole body dose). Those criteria where appropriate can also be expressed in terms of corresponding annual effluent discharge limits authorised by National authorities.

Figure 3.2 : Severity scale for rating incidents and accidents in nuclear installations other than reactors (France)

LEVEL	DEFINITION	CRITERIA	EXAMPLES
5	ACCIDENTS AFFECTING THE PUBLIC	<ul style="list-style-type: none"> <li>Off-site releases of radioactive products of the same order of magnitude as the authorized annual limits or large scale releases of associated dangerous chemical products, involving significant health risks for the public.</li> <li>Major damage of active parts of the installation.</li> <li>Exposure of workers to ionizing radiation or associated dangerous chemical products, to a degree requiring specialized medical care.</li> </ul>	<ul style="list-style-type: none"> <li>Fire in a silo La Hague - 6.1.1981 (cf. Bulletin SN n° 19)</li> <li>Accidental discharge of UF6 Comurhex - 1.7.1977</li> </ul>
4	ACCIDENTS IN THE INSTALLATION	<ul style="list-style-type: none"> <li>Off-site releases of radioactive products of the same order of magnitude as the authorized annual limits or large scale releases of associated dangerous chemical products, involving no significant health risks for the public.</li> <li>Major damage of active parts of the installation.</li> <li>Exposure of workers to ionizing radiation or associated dangerous chemical products, to a degree requiring specialized medical care.</li> </ul>	<ul style="list-style-type: none"> <li>Fire in a silo La Hague - 6.1.1981 (cf. Bulletin SN n° 19)</li> <li>Accidental discharge of UF6 Comurhex - 1.7.1977</li> </ul>
3	INCIDENTS AFFECTING SAFETY	<ul style="list-style-type: none"> <li>Release outside the site of an amount of radioactive substances greater than or equal to one tenth of the authorized annual limits or limited amounts of associated dangerous chemical products.</li> <li>Internal leaks of radioactivity resulting in heavy contamination of an area normally accessible to staff.</li> <li>Degraded condition of containment systems or safety systems.</li> <li>Exposure of workers to ionizing radiation of more than the authorized limit or associated dangerous chemical products to a degree requiring placing under medical observation.</li> </ul>	<ul style="list-style-type: none"> <li>Accidental discharge of LFC, Eurodif - 10.8.1985 (cf. Bulletin SN n° 47)</li> <li>Loss of electrical power supplies La Hague - 15.4.1988 (cf. Bulletin SN n° 44)</li> <li>External exposure of staff La Hague - 20.5.1986 (cf. Bulletin SN n° 37)</li> </ul>
2	INCIDENTS AFFECTING THE INSTALLATION	<ul style="list-style-type: none"> <li>Release outside the site of an amount of radioactive substances greater than or equal to one tenth of the authorized annual limits or limited amounts of associated dangerous chemical products.</li> <li>Internal leaks of radioactivity resulting in heavy contamination of an area normally accessible to staff.</li> <li>Degraded condition of containment systems or safety systems.</li> <li>Exposure of workers to ionizing radiation of more than the authorized limit or associated dangerous chemical products to a degree requiring placing under medical observation.</li> </ul>	<ul style="list-style-type: none"> <li>Accidental discharge of LFC, Eurodif - 10.8.1985 (cf. Bulletin SN n° 47)</li> <li>Loss of electrical power supplies La Hague - 15.4.1988 (cf. Bulletin SN n° 44)</li> <li>External exposure of staff La Hague - 20.5.1986 (cf. Bulletin SN n° 37)</li> </ul>
1	INCIDENTS AFFECTING THE INSTALLATION	<ul style="list-style-type: none"> <li>Release outside the site of an amount of radioactive substances greater than or equal to one tenth of the authorized annual limits or limited amounts of associated dangerous chemical products.</li> <li>Internal leaks of radioactivity resulting in heavy contamination of an area normally accessible to staff.</li> <li>Degraded condition of containment systems or safety systems.</li> <li>Exposure of workers to ionizing radiation of more than the authorized limit or associated dangerous chemical products to a degree requiring placing under medical observation.</li> </ul>	<ul style="list-style-type: none"> <li>Accidental discharge of LFC, Eurodif - 10.8.1985 (cf. Bulletin SN n° 47)</li> <li>Loss of electrical power supplies La Hague - 15.4.1988 (cf. Bulletin SN n° 44)</li> <li>External exposure of staff La Hague - 20.5.1986 (cf. Bulletin SN n° 37)</li> </ul>

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## 4. General Safety

### 4.1 Internal safety issues

In most of the nuclear fuel cycle facilities, the radioactive materials are generally in an easily transportable and dispersable form, as a gas, or in solution or as a finely divided powder. However, the temperatures and pressures typically encountered are substantially lower than in nuclear reactors and, as the plant is operating in nuclear sub-critical conditions, deviations are therefore unlikely to develop rapidly into dangerous situations. Nevertheless the radioactive materials encountered in fuel cycle facilities may be very diverse and in a number of chemical and physical forms; they are often in connection with flammable or chemically reactive substances as part of the process. While automation has increased, operators are often still intimately concerned with management and control of processes. The interaction and super-position of chemical, mechanical and nuclear hazards leads to some very specific potential hazard scenarios, from which man and the environment are protected by good design and construction of facilities and by their safe operation.

#### 4.1.1 Containment and ventilation

The levels of radioactivity in fuel cycle installations vary depending on the radionuclides concerned; preventative measures however, are always taken to reduce the hazard to workers associated with ingestion or inhalation, and hence the use of containment structures and a ventilation system. In some cases, there is the added risk of chemical toxicity (from uranium), which may even become the dominant potential hazard, as in the case of handling  $UF_6$ .

The uncontrolled dispersion of radioactive substances to the environment from accidents can only occur if the successive containment barriers, separating these substances from the environment, are all impaired. A containment barrier may be afforded by:

- the process equipment itself;
- the surrounding cell or glove box;
- the room or building in which the plant is located.

Radiotoxic radionuclides (plutonium, fission products) are handled in leak-proof cells located, wherever possible, inside a second containment. All these facilities are housed in a building which acts as the final containment barrier to the environment.

Containments cannot be absolutely leak-tight in all circumstances, either due to technical impossibilities (large concrete-shielded cells in reprocessing plants) or because certain operations might involve temporary leak-tightness failures, for instance when equipment or products are being introduced, or removed, or even in the event of an incident such as a pierced glove during glove box operations. Containment systems are therefore augmented with dynamic containment supplied by means of a ventilation system, designed so that the pressure inside the containment is always lower

than outside. In this way a cascade of reducing absolute pressure is established between the air outside, inside the building and inside the different containments so that in the event of a leak, radioactive particles or gases cannot flow out.

Ventilation of these containments, with discharge of exhaust gases through a stack via a gas-cleaning process (e.g. scrubbers, chemical traps, high-efficiency filters and electrostatic precipitators), can reduce environmental discharges of radioactive materials to very low levels. In such cases the ventilation system might also be regarded as a containment barrier.

#### 4.1.1.1 *Containments*

Primary containments vary in type depending on the products involved and the radiotoxicity of the radionuclides present. They are located as close as possible to the radioactive materials handled. In some cases, especially for liquid phase operations, primary containment may be provided by the equipment used for the process.

##### *Uranium enrichment*

Regardless of whether the process is gaseous diffusion or ultracentrifugation, containment is provided by the equipment, which is highly leak-tight as a fabrication quality requirement. Secondary containment consists of the cells housing the most sensitive systems, or the building itself, so that in the event of an accident release into the environment is kept to a minimum. It should be noted in this connection that the dominant hazard arises from the toxicity of the HF gas produced through hydrolysis of  $UF_6$  in contact with air moisture but normally, adequate preventive measures are taken into account in the design of a facility so as to minimise the risk.

##### *Uranium fuel fabrication*

At  $UO_2$  fuel fabrication facilities, containment is required for the furnaces converting  $UF_6$  to  $UO_2$ , and also wherever  $UO_2$  is present in powder form (crushers, mixers and compacting presses). The latter equipment is not leak-tight and systems are installed to draw off dust and aerosols to keep atmospheric contamination in the buildings within acceptable limits.

##### *$PuO_2$ or MOX fuel fabrication*

Since plutonium is a highly radiotoxic element, all operations are carried out in highly leak-tight glove boxes. The glove box itself becomes the containment barrier for operators.

##### *Reprocessing*

Owing to the large amount of radionuclides in spent fuel (plutonium, fission products and activation products), all operations are conducted in containments.

For liquid phase operations (fuel dissolution, plutonium, uranium and fission products separation) the equipment acts as the first containment barrier. It is made of materials of suitable composition (stainless steel, zirconium) and thickness for the physical and chemical operating conditions, to minimise the risk of failure. Special attention is paid to the design and quality of equipment, especially relating to welds. In the event of a leak, solutions are recovered in secondary containment to prevent any radioactive liquids from contaminating the ground.

The equipment is installed in shielded cells which act as the second containment barrier and also protect personnel against gamma and neutron radiation.

For dry operations, the equipment cannot always provide the first containment barrier. Such is the case in fuel element shearing, which generates gases and aerosols. The shielded cell is the first containment barrier in that situation. Cell contamination therefore has to be kept to a minimum. This is achieved by using suitable ventilation systems picking up most of the gases and aerosols produced at source. Likewise, operations leading to the production of PuO<sub>2</sub> powder require precautions to ensure that the contamination of glove boxes is minimised since they then act as the first containment barrier.

#### 4.1.1.2 Ventilation

Nuclear installations are equipped with a ventilation system which serves to provide several safety functions, these include:

- Dynamic containment to counteract any leak-tightness failures of the static containment in the different envelopes. Ventilation provides a cascade of reducing absolute pressure to ensure that air flows from the buildings where personnel are present, towards the hot cells and the containments housing radioactive materials.
- Treatment of the gaseous effluents removed from the equipment and, where applicable, treatment of the air drawn from the buildings by routing the gases and air to suitable apparatus (such as filters, traps) before being released into the environment. Gaseous effluents are monitored.
- Renewal of air in the buildings where personnel are present.
- Monitoring the ventilation air for contamination serves as a warning mechanism for the identification of breaches of containment.

The level of importance placed on the safety functions of ventilation systems is evaluated for each type of installation and application. For the most sensitive installations, dynamic containment might be necessary on a permanent basis. In this case the ventilation system is designed for an extremely low failure rate probability (back-up power supply, equipment redundancy). Monitoring systems, checks and periodical testing as well as maintenance procedures are provided.

Proper ventilation system design maintains an effective dynamic containment for all plant operating conditions, including accident situations considered to be plausible. Special attention is also paid to the fire risk by ensuring that:

- the ventilation system does not worsen this risk, for instance by leaving scope for fire to spread through the ventilation ducts;
- where possible, ways of controlling any fire using the ventilation system itself are available (presence of fireproof shutters, pre-established ventilation procedure during a fire);
- where possible, equipment is fireproof;
- the last filtration stage prior to release into the environment remains sufficiently effective not to lead to unacceptable releases (filter resistant to higher temperatures, suitable location in the ventilation system).

## 4.1.2 *Criticality*

### 4.1.2.1 *General*

All operations with fissile materials outside nuclear reactors are performed in such a way as to prevent the establishment of nuclear chain reactions and the concomitant release of energy. Though the energy would manifest itself principally as heat in the particular medium, the associated ionizing radiations might be lethal to nearby personnel without their being aware that an incident has occurred, or without any apparent damage being sustained by the process equipment; the only indication of any untoward occurrence may be the sounding of radiation alarms. Additionally, damage to equipment could be caused resulting in the possible escape of fission products generated by the incident and perhaps additional radioactive materials, particularly if irradiated materials were being processed. Damaged equipment would result in the interruption of operational schedules and could lead to the dispersal of sufficient active material to present a severe decontamination problem or an environmental hazard.

The achievement of criticality depends upon:

- the properties of the fissile material;
- the mass of fissile material present and its distribution among the components of the system being assessed;
- the mass and distribution of all other materials associated with fissile material.

Methods of control of criticality safety in any process include, but are not limited to, any one or a combination of the following:

- control of the mass of fissile material present in a process;
- control of the geometry (limitation of the dimensions or shape) of processing equipment;
- control of the concentration of fissile material in solutions;
- the presence of appropriate neutron absorbers.

The control of criticality safety by such methods as those indicated above can be achieved by:

- equipment design;
- use of process control instrumentation;
- compliance with operating procedures.

Where possible, the maintenance of control depends either on safety features incorporated in the equipment or on the instrumentation rather than on administrative control. This topic is also discussed in section 5.2.2.

### 4.1.2.2 *Factors affecting criticality*

A number of factors should be considered both singly and in combination for a proper analysis of criticality safety. Some of the more important factors are:

### *Moderation*

The presence of neutron moderating material mixed with fissile material can substantially reduce the mass of fissile material necessary to achieve criticality. Water, oil and similar hydrogenous substances are the most common moderators present in the handling and the processing of fissile material.

### *Reflection*

The most effective neutron reflector commonly encountered in handling and in processing fissile material is water of thickness sufficient to yield maximum nuclear reactivity. However, careful consideration should be given to systems where significant thicknesses of other common structural materials (e.g. wood, concrete, steel), which may be more effective neutron reflectors than water, may surround or cover an appreciable fraction of a container surface.

Where the extent of neutron reflection by such partial or potential reflectors as vessel walls, supports, or personnel cannot be assessed accurately, or the possibility of neutron reflecting materials being introduced accidentally by flood or fire-fighting cannot be discounted, the most effective reflector conditions should be assumed in all assessments. If limitation of reflecting materials can be guaranteed, appropriate relaxations are permitted.

### *Interaction*

Consideration should be given to neutron interaction between units when more than one unit containing fissile material is present. It is possible to reduce neutron interaction to acceptable proportions either by spacing items or by insertion of suitable neutron moderating and absorbing materials between items, or by some combination of these methods.

### *Neutron absorbers*

Equipment and processes can conform to the requirements of criticality safety by the use of neutron absorbing materials, such as cadmium and boron, providing their suitability has been confirmed and their presence can be assured. The use of solid neutron absorbers in the construction and assembly of equipment is preferred; the use of solutions of neutron absorbers is less desirable because of the procedures and controls necessary to ensure their presence. Neutron absorbing materials are most effective for neutrons of thermal energy. However, care should be exercised to ensure that their effectiveness is not seriously reduced in operational or accident conditions which might change the fissile assembly into one characterised by neutrons of intermediate or high energy.

#### *4.1.2.3 Possible abnormalities*

The effect of the occurrence of possible abnormal conditions should be considered in the assessment of safety. These include such factors as:

- loss or introduction of moderating material into or between units of fissile material, e.g. evaporation, precipitation, dilution and flooding;
- introduction of neutron reflecting material near units of fissile material;
- change of shape of fissile material due to such occurrences as vessel leakage or breakage;
- change in operating conditions, e.g. loss of flow, precipitation, excessive evaporation, violation of mass limits;

- change in conditions of neutron interaction, e.g. collapse or overturn of equipment, passage of fissile material in transport.

#### 4.1.2.4 *Assessment philosophy*

Wherever possible the specification for criticality safety should be established on bases derived directly from experiments. In the absence of directly applicable experimental measurements, the results of calculations are acceptable provided they are shown to compare favourably with experimental data.

In all specifications the margin of safety should be commensurate with the uncertainty in the basis of the assessment, the probability of its violation, and the seriousness of the consequences of a conceivable criticality accident.

Operations should, in general, incorporate sufficient safety features to ensure that two independent concurrent changes must occur in the conditions originally specified as essential to criticality safety before the system may become critical. Following any actual occurrence of one of these changes the safety of the process should be re-evaluated.

#### 4.1.2.5 *Criticality accident*

The probability of a criticality accident is inherently extremely low in plants processing only low-enriched uranium and/or mixed plutonium-uranium oxide in dry solid form. It is higher in other plants, especially those in which uranium and plutonium are present in solution.

Criticality accident tests carried out in France with highly enriched uranium solutions (CRAC and SILENE programmes) and accidents that have actually occurred in the United States and the United Kingdom facilities show that an accident leading to  $5 \times 10^{18}$  fissions, corresponding to a total energy of 40 kWh, is a reasonable envelope hypothesis. However, greater yields are conceivable for large volumes of solution. A critical excursion of  $5 \times 10^{18}$  fissions could not have serious physical consequences and its effects would be negligible beyond the site boundary. In those parts of a plant where there is no adequate radiation shielding, a reliable criticality accident detection system would be installed. Escape routes and emergency procedures are pre-planned and regular emergency drills practised by the staff.

A summary of criticality accidents which have happened in fuel cycle plants is given in section 7.3.

### 4.1.3 *Fire hazards – general*

The safe use of radioactive substances requires barriers to ensure adequate protection for workers and the environment. The purpose of these barriers is to provide either (static or dynamic) containment for potential contaminants or biological shielding against radiation hazards. Their efficiency, however, may be reduced or even neutralised in the event of a fire. Requisite measures should therefore be taken to ensure that the likelihood of such an event is minimised.

Practices used by OECD countries to deal with fire hazards differ considerably. The following sections deal with some general topics based mainly on French technology.

In comparison with nuclear power plants, where a major effort has been made to standardize protection procedures, the hazards encountered in laboratories and factories vary considerably and a special fire hazard analysis should be carried out for each individual installation. With older installations, analyses of this kind allow safety authorities to determine what improvements are needed to meet current safety standards. In order to carry out these analyses, some Member countries have developed special design codes and expert systems which they use in conjunction with existing technical rules and regulatory requirements or guides.

The analysis of fire hazards involves identifying the causes of fires, assessing their potential consequences and, where possible, estimating the probability of such fires occurring. However, although these estimates can provide valuable information on which to base decisions or to identify weaknesses that might otherwise have gone undetected, they need to be used with caution. Certain protective measures might have to be taken, even though the probability of fire occurring may seem low, whenever there is a possibility that a fire may have significant consequences with regard to nuclear safety.

Analysis of fire hazards also involves a sequential review of the provisions made for preventing, detecting and fighting fires.

#### *Fire hazard analysis*

A distinction needs to be made between "conventional" hazards commonly encountered in industry, such as those involving electrical equipment, which are well known and which depend in particular on the fire load, and those relating specifically to a given process used at an installation.

Examples of the latter in reprocessing plants include:

- the risks arising from the presence of pyrophoric zircaloy fines in irradiated fuel element chopping/leaching operations;
- the risks associated with the use of the flammable solvents utilised in the Purex process in irradiated fuel reprocessing.

#### *Fire preventive measures*

Prevention is undoubtedly the single most important aspect of fire protection. Facilities should be designed to limit fire risks (probability of occurrence multiplied by the consequence) by taking measures to ensure that fires do not break out and that, were a fire to break out despite the precautions taken, its consequences would be limited. A number of both general and specific measures are taken to accomplish this two-fold aim, consisting principally of:

- minimising the fire load of individual rooms;
- choosing materials according to functional criteria and fire-resistance ratings; thus, building-components and fittings are made from materials considered to be non-combustible;
- compartmentalising buildings as far as possible in order to prevent fires spreading. Buildings should be divided into fire areas. Should a fire break out within a given fire area, it must not be able to spread beyond the sector. The higher the fire risk, the greater the number of areas a building should have.

One very important aspect of fire hazard analysis is determining which rooms require special consideration. Rooms are categorised under three main headings:

- rooms with high fire loads;
- rooms housing safety-related items of equipment. Items of equipment that fall under this heading are generally those associated with a function that is important to safety, i.e. items whose degradation might have radiological consequences considered to be unacceptable;
- rooms whose degradation might have direct or indirect radiological consequences that would be considered to be unacceptable. In rooms in which radionuclide activity levels are high, there is a constant risk of radioactivity being released.

One item that requires particular attention is the design of ventilation systems. Dynamic containment comprises ventilation ducts and filter units which, unless they are of suitable design, may constitute weak points in the system.

In some situations, fire dampers are mounted in the ventilation system to close automatically on receipt of a signal from the fire-detection system.

Electrical cable specifications must ensure that they cannot provide a path along which fires can spread and that they release minimal amounts of smoke and toxic or corrosive gases when exposed to fire.

#### *Fire-detection systems*

Analysis of fire hazards will identify the rooms and containments that need to be fitted with automatic fire-detection systems. Fire-detectors are fitted by an approved installer and in accordance with well-defined rules. The alarm is relayed to locations where help may be summoned immediately (facility control room, emergency team command post).

#### *Fire-fighting resources*

Compared with conventional industries, the nuclear industry poses certain constraints in this respect:

- difficulty or impossibility of gaining access to the fire (danger of irradiation, special clothing for fire fighters);
- the need to maintain a certain level of containment (in order to limit releases of radioactivity to the environment);
- restrictions in terms of the extinguishing agents that may be used.

Thus in the case of accidents involving a risk of criticality, the use of water may be prohibited.

#### *Periodic inspection – safety instructions*

To ensure efficiency and operability, the above-mentioned fire protection systems may call for the introduction of a number of procedures such as:

- periodic testing, inspection and maintenance of the devices associated with fire protection systems (extinguishers, fire dampers, fire detectors);
- general and detailed orders for fire fighters;
- fire-fighting plans;

- fire drills;
- training for operating staff and emergency teams.

## **4.2 External safety issues**

The safe operation of nuclear fuel cycle facilities can be affected by events which occur externally to the building or site. These "external hazards" can be man-induced or can result from natural phenomena. Hence, to ensure that both site personnel and the general public are adequately protected, nuclear fuel cycle facilities are required to be able to cope with reasonably foreseeable events of this type. The designers of new facilities are required to consider the effects of earthquakes, fires and explosions, aircraft crashes, extreme weather conditions and flooding. The following section addresses each of these events in turn and gives an indication of the approach and standards adopted by various countries.

### **4.2.1 Seismic events**

Where the radioactive inventory or the consequences of loss of containment are such that site personnel or the general public would be significantly affected by the damage done to a nuclear fuel cycle facility as a result of an earthquake, plants are seismically qualified. The extent of seismic qualification depends upon the type of plant or structure and its required performance. In order to determine how a plant will respond to a particular seismic event it is necessary to know the magnitude of the disturbance, and to relate the magnitude to the "intensity" (severity at the site from an event occurring away from the site).

Most countries operating nuclear fuel cycle facilities adopt a deterministic approach to seismic assessment. This approach ideally requires the evaluation of the seismology and geology in the area of the plant in order to determine the relationship between the magnitude or intensity of the event and its frequency of occurrence. Generally, however, one can only calculate probabilities that a given intensity will be exceeded. Once this relationship has been derived the licensee can specify the design basis event (DBE) (guidelines may be issued by the safety authorities). The use of  $10^{-4}$  probability of exceedance has been widely accepted for determining the DBE in the United Kingdom for instance. Designers use this DBE to design the plant and evaluate the safety performance of the plant.

The above approach is used in most countries although there are some national differences in detailed application.

### **4.2.2 External fires and explosions**

For safe operation, a nuclear fuel cycle facility is either designed to cope with external fires or explosions, or it is shown that the probability of occurrence of an event which would prejudice safety is acceptably low.

The sources of external fires and explosions are such things as the presence of petrochemical installations in the vicinity of the nuclear site, the proximity of forests, and of road, rail or sea routes where there is the transport of natural gas, and the routing of gas or oil pipelines.

In order to demonstrate that the risks associated with these external hazards are acceptable the licensee is required to identify all potential sources of hazard and then calculate the risk of damage. The radiological consequences of any damage are evaluated and shown to be within acceptance criteria. In the United Kingdom the licensees are required to carry out a survey of potentially hazardous installations and transport operations concerning hazardous materials. Typically a radius of 10 km from the installation is chosen for this survey. The risks are evaluated in the case of explosions against over-pressure criteria. Flammable liquids and missiles resulting from explosions are assessed against distance from the plant, and hence their potential for physical damage. Toxic hazards are assessed against specific gas concentrations.

If the probability of a fire or explosion causing damage to a nuclear facility is above acceptable limits, then design measures will need to be taken to limit the damage and hence radiological release. If the probability is calculated to be sufficiently low, then no design measure will be required. For operating plants the hazard posed by external factors is monitored throughout the lifetime of the plant. The construction of facilities adjacent to a nuclear site which could prejudice the safety of that plant is usually controlled via land-use planning requirements.

#### **4.2.3 Aircraft crash hazard**

An aircraft crashing onto a nuclear facility represents a man-made external hazard, and the risks associated with this type of incident are required to be evaluated. The approach used to evaluate this hazard is similar in most countries. The potential for aircraft crash is evaluated for the site for large and small commercial aircraft and military jets. In the United Kingdom aircraft crash statistics are used to determine the crash probability in the vicinity of the nuclear site; the probability of impact onto a nuclear facility is then calculated by evaluating the vulnerable impact area. This probability is compared with the criteria for a major release of radioactivity of  $10^{-7}$  per year. If the calculated impact probability is less than this no specific measures are required for aircraft crash. However if the impact probability is greater than  $10^{-7}$  per year the designer is required to introduce measures to mitigate the consequences of such an event so that radiological releases are within acceptable criteria. In evaluating the consequences of impact or the adequacy of the design to resist aircraft impact, realistic crash scenarios are often considered. These require knowledge of such factors as the angle of impact and the potential for fire and explosion resulting from the aviation fuel.

To minimise the potential for aircraft impact, aircraft are excluded from airspace above nuclear fuel cycle facilities; for example, in the United Kingdom there is an overflying restriction zone.

When considering the probabilities of aircraft crash the situations differ from one country to another. For example, French military aircraft crashes occur, in general, close to air bases whereas in the United Kingdom such crashes are more widely distributed across the country, not necessarily only in training areas. It is also important to take site-specific circumstances into account. For example: at Cadarache in France, flights of forest fire-fighting aircraft are considered. It is also necessary to consider realistic crash scenarios. For example, the angle of impact is normally taken into consideration in France.

When it comes to translating an aircraft impact into a given loading on the structure, Germany is the only country to have developed a specific loading function. This approximates the loading curve believed to result from a Phantom military aircraft flying at approximately 215 meters per second.

It was noted that fire could not in general be ruled out following an aircraft crash (most crashes are followed by fire) and that this sometimes leads to a specific fire protection requirement. For example, in a Belgian assessment five tonnes of aircraft fuel burning for three minutes was taken as a typical case.

It is felt that these measures, when taken, provide not only aircraft protection but a form of "umbrella" protection against a variety of other hazards such as a terrorist attack. Given a defined aircraft threat in terms of either a specified aeroplane or loading, at least approximate methods are available to assess the effects on a structure. Germany, the United Kingdom and the United States use specific calculational codes or formulae.

#### **4.2.4 *Extreme weather conditions***

Extreme weather conditions can cause an external hazard to the safe operation of a nuclear facility, either by directly affecting the integrity of the structure or through indirect effects such as bringing down power lines or affecting cooling water supplies. The general approach is to use a deterministic, design basis value for the condition and assess the impact of such an event on the operational capability of the plant. In the United Kingdom for example the design basis values for assessment are chosen as those associated with the 10,000 year return period event.

The impact and hence the design provisions vary depending upon the type of hazard. For example extreme wind loadings are associated with rapid structural loading and thus design provisions are required in the same way as other potential rapid loading events such as earthquakes. However, the effects of extreme precipitation or extreme temperatures would take time to develop and hence there is time for operational actions to be taken to limit the consequences of the event.

Typically, the extreme weather conditions used to design and/or evaluate fuel cycle facility response are wind loadings, tornadoes, rainfall, snowfall, temperatures and flooding.

##### *Tornadoes*

Protection against tornadoes depends on the meteorological conditions of the area where the nuclear facility is installed. Since strong tornadoes occur in some areas of the United States it has special regulations with regard to tornadoes. The country is subdivided into three distinct sections with maximum wind speeds of 384, 480 and 576 km per hour. These extreme wind speeds have only a locally devastating effect but require special construction designs. Experimental evidence has been provided to show that steep pressure changes may damage the integrity of ventilation systems which have not been designed to withstand such conditions. The Federal Code 10 CFR 50 prescribes the design criteria with which nuclear facilities must comply. Although hurricanes with wind speeds of 160 km per hour have great climatic influence, they do not provide an additional threat to conventionally well-designed buildings.

In the United States, all types of extreme weather conditions are considered by the designers of higher potential risk non-reactor fuel facilities, such as reprocessing plants or plutonium recycle plants, depending upon the location in the country. Tornadoes are frequent in the midwest and hurricanes can be severe along the eastern coastline. Both wind phenomena are capable of lifting and propelling objects such as automobiles or telephone poles. These missiles are taken into consideration during the design stage, both for their initial impacts and possible secondary fragments spalled from concrete walls or other types of momentum transfer.

Analytical methods have been developed to determine the effects of sudden pressure changes on facilities, primarily as connected with tornadoes. A high-pressure differential can create a significant external or internal force on building walls, as well as create high-velocity airflows in either forward or reverse direction. Some analyses have coupled earthquake damage followed by relatively strong attendant winds in order to gain a perspective of design resistance to such extreme events. The analytical techniques have been used to estimate both the quantities and location of radioactive material movement through and from processing cells. Maximum credible doses to individuals, both on the plant site and beyond its controlled boundaries have been calculated by such techniques.

Maximum water flooding conditions connected with large storms have also been estimated.

The design against earthquakes is generally considered as an umbrella protection for other types of external hazards.

#### *Extreme temperatures*

The duration of extreme temperatures has to be considered in order to assess the effect of freezing, for example, of secondary equipments (cooling circuits) of fuel cycle plants.

#### *Snow*

The occurrence of snow and its effects have to be taken into account in the safety design and analysis.

#### *Floods*

Floods have to be taken into account when designing a facility. Two approaches to cope with this type of hazard have been put forward:

- some countries take into account the highest flood levels historically recorded and erect the nuclear facilities on specific locations or at sufficient elevation to avoid major damage;
- some other countries, where dams are widespread and built up-stream of potential or existing nuclear sites, take into consideration this hazard by designing the buildings in such a manner that they withstand the waterwave released from the collapsing dam.

## 5. The Safety of the Individual Stages of the Nuclear Fuel Cycle

### 5.1 Front end of the fuel cycle

#### 5.1.1 *Uranium mining and milling*

The mining and milling of uranium do not give rise to safety problems of a nature or type which are associated with the operations of nuclear power reactors and other fuel cycle facilities. However, in addition to the various occupational and environmental risks which are associated with any type of mining and milling operation, uranium mining and milling present certain radiological protection concerns because of the ionizing radiation emitted by the radionuclides in uranium ores.

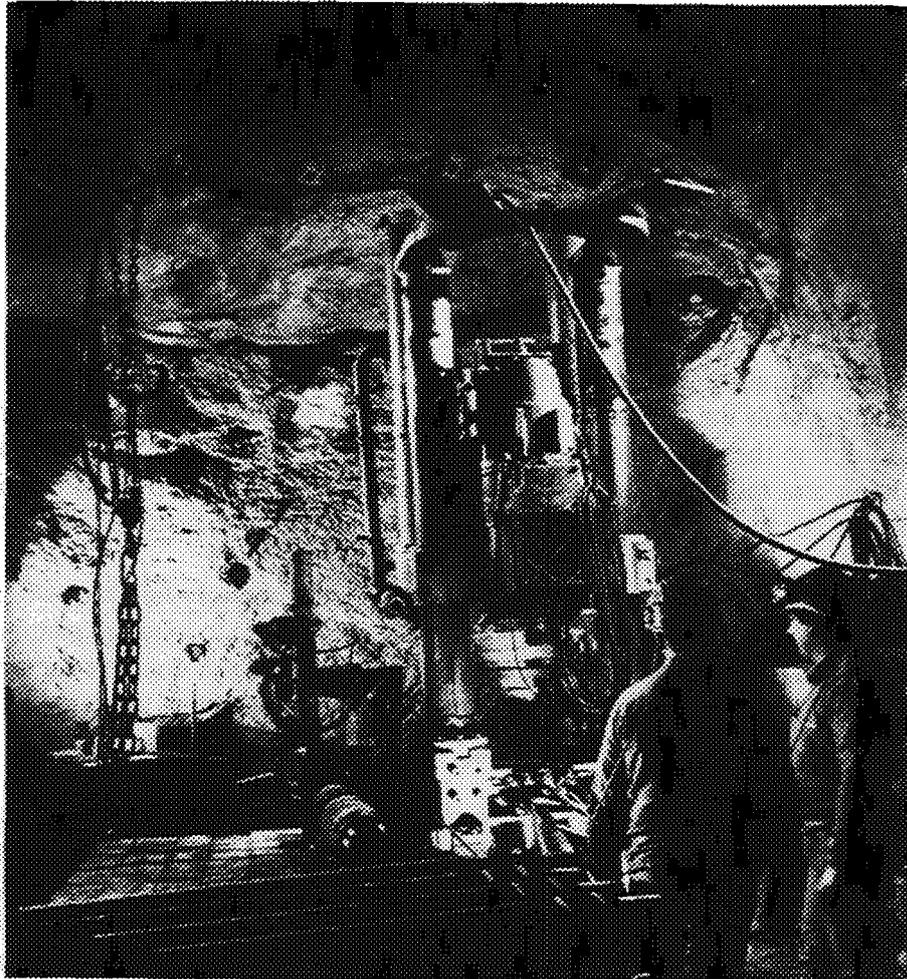
The safety concerns with uranium mining and milling generally fall into two categories: the first involves the protection of workers against the risks of exposure to external gamma radiation, the inhalation of radon daughters, and the inhalation of long-lived alpha emitters (uranium dust); the second concerns the protection of the public and the environment from the effects of the uranium mill tailings (Figure 5.1).

The protection of uranium miners, who are routinely among the most highly exposed workers in the nuclear fuel cycle, is achieved in underground mines mainly by controlling ventilation rates. The short-lived radon daughter contamination is cleared from the air of the work areas before unacceptable levels are reached. In high-grade mines it is often necessary to shield the operators or have the equipment operated by remote control in order to maintain worker exposures below the limits established for external gamma exposure. Protection against radioactive dust is achieved basically by wet drilling and by keeping the stoped ore very moist.

Historically, open-pit mining generally has caused the least amount of concern with the radiological protection of workers, whereas in underground mining facilities both the external gamma radiation and the inhalation pathway need to be closely monitored. A well-operated open-pit uranium mining operation should not expose workers to any significant radiation dose, while underground mining operations with ore grades in the order of 0.1 per cent give rise to individual average annual exposures of external gamma dose and radon daughter exposure (1) (Figure 5.2)

Management of mill tailings (the finely ground remains of the ore after the uranium has been removed) during the operating phase presents special problems. The mass of mill tailings corresponds to that of the processed ore, and thus amounts to many millions of tonnes per year worldwide. Tailings contain most of the original radioactivity of the ore, although their specific activity is minimal compared to many other low level radioactive waste products. They also contain a level of background radiation – the daughters of thorium-230 (six alpha emitters and four beta emitters). Because of this relatively low specific activity the tailings cannot result in exposure situations that would produce immediate health effects. However, a major release of tailings could

contaminate a large area of the environment and, because of the radionuclides presents (particularly radium-226, radon-222 and lead-210), could contribute measurably to the level of background radiation.



**Figure 5.1 View of drilling in a mine**

*(Source: Cogéma - France)*

Accidental releases resulting from the failure of tailings retention dams have been reported at a significantly higher relative frequency than from failure of water retention dams (2). However such releases, sometimes involving large volumes of liquids and solids, have not resulted in noticeable radiological consequences. The main causes of dam failure and pipeline rupture are engineering and construction defects. It is for this reason, among others, that high standards are now required in the design, construction and operation of retention facilities for uranium mill tailings. Also for this reason, it is desired where possible, that tailings be stored below the surface level in basins dug for the purpose, or in old open-cast mines, in a manner that does not alter the original topography nor allow the movement of radioactive sludge. Ideally, the bottom of such structures would remain permanently higher than the water table.

A survey performed at the request of the US Department of Energy (DOE) involved a thorough examination of all aspects of radiation protection in the United States with regard to communities in the vicinity of uranium mill tailings (200,000,000 tonnes of tailings at 51 sites,

covering a total of 1,300 hectares) (6). In the summary of the conclusions to this report, the working group responsible for the survey stressed certain unavoidable aspects and characteristics of this kind of radioactive waste which make it difficult to transfer directly, to tailings management, the doctrine and rules that have been adopted for the safe handling of radioactive wastes from the rest of the fuel cycle. One of these characteristics is that one of the radionuclides contained in the tailings, radon-222, has the properties of an inert gas and, as such, tends to escape from solid matrices where it is produced. It then passes easily through confinement barriers and, via the air pathway, to the nearest communities. An analogous situation is when reactor wastes contain tritium ( $H_3$ ), which is also a gas and is therefore very difficult to confine. The difference, however, is that  $H_3$  decays according to its own half-life of 12.3 years, whereas the radon-222 activity in tailings decays with the half-life of thorium-230, which is  $8 \times 10^4$  years. Thus, the radon problem cannot be resolved through decay, in the way it can be done for tritium-containing waste or most waste containing gamma-emitting artificial radionuclides (such as caesium and radioactive strontium) within, say, 300 years. The report in question identifies the radon-222 as the primary hazard and recognises it as significant for communities living in the immediate vicinity of the site boundaries, certainly if they are less than one kilometer away.



**Figure 5.2 View of open mine**

*(Source: Rabbit Lake - Canada)*

Another aspect of uranium tailings management is that all the radionuclides involved are naturally occurring. Whereas, due to the absence of "background noise", artificial radionuclides which have escaped into the environment can be detected even after having been highly diluted while travelling very long distances on a worldwide scale. But it is almost impossible, if more than a few kilometers away from their point of release, to identify radionuclides as originating from a given tailings from mining waste, whether the pathway concerned be air or water.

The above-mentioned report (5) recommends that when determining management strategies, account should be taken of the special characteristics of the site concerned. The use of generic impact assessment models is not recommended especially as regards the dispersion of radon over a short distance, for which it is very difficult to give an accurate enough figure, even though the phenomena involved are very clearly identified at a qualitative level.

Populations living in uranium-bearing areas are usually exposed to radiation through radon in the home, or sometimes through the consumption of local radium-containing water, to annual dose equivalents which are significant compared to the level considered as a maximum for populations living close to sites for the storage of artificial radioactive waste.

In recent years, countries mining and milling uranium have paid greater attention to the regulation, management and operation of facilities for the disposal of waste and mill tailings. Disposal of mill tailings is outside the scope of this report.

### 5.1.2 *Uranium refining and conversion to uranium hexafluoride*

The milling process produces a chemically impure form of  $U_3O_8$ . This product must undergo further processing to produce uranium dioxide which can be made into fuel or which can be converted to uranium hexafluoride ( $UF_6$ ). The refining operation is an enhanced version of the dissolution and extraction operations which are conducted at the milling facilities. The hazards to the workers and the public associated with the refining operations are not different from those associated with the chemical operations at a mill (Figure 5.3).

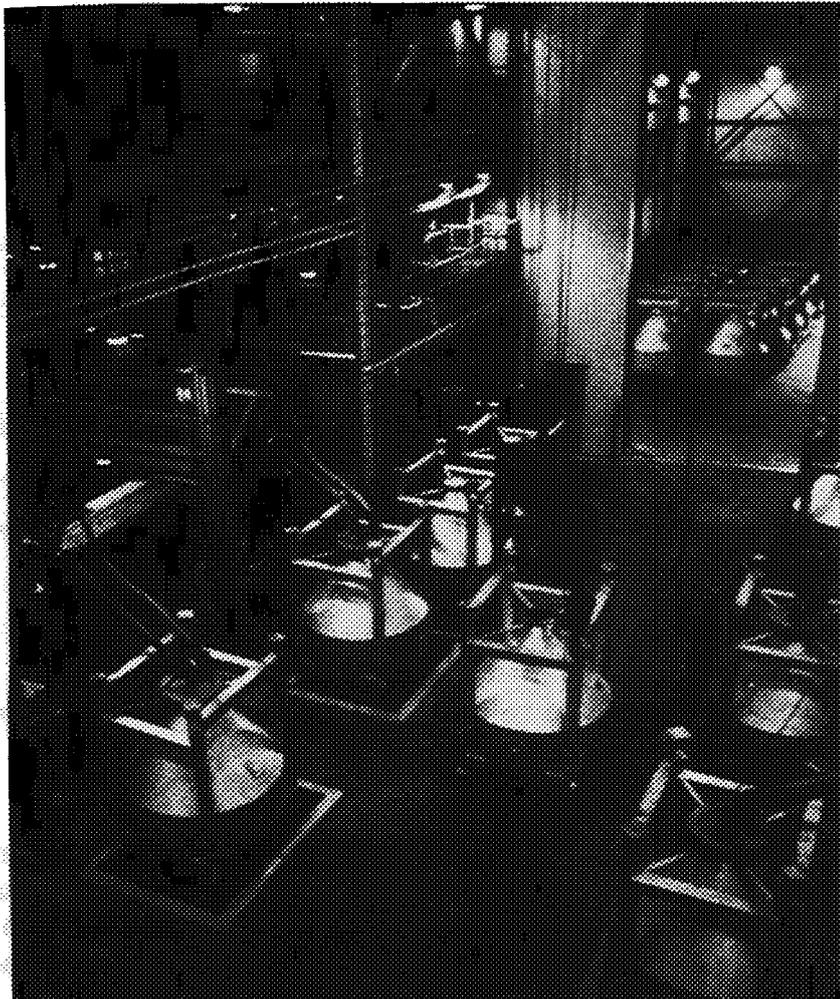


**Figure 5.3 Uranium refining plant—"Yellow-cake" production**

*(Source: Blind River Facility – Canada)*

The production of  $UF_6$  has been carried out since the early 1940s in the United States, since 1952 in the United Kingdom and France, and since 1969 in Canada. The design and operation of the facilities in this part of the fuel cycle require that not only are the radiological concerns of processing uranium satisfactorily addressed but that the main processing chemicals, HF and  $F_2$ , be safely handled. Most plants are designed to protect against the hazardous material risk discussed below.

The CSNI Specialist Meeting on the safety problems associated with the handling and storage of  $UF_6$  held in Boekelo in 1978 concluded that the existing processes for uranium refining and conversion to  $UF_6$  give rise to no significant radiological hazards, and that the safety problems associated with these operations are essentially those of a conventional chemical industry dealing with toxic chemicals. However, adequate precautions are implemented to protect workers from exposures to high radiation fields associated with the flame reactor and ash handling operations, and from the hazards of inhaling uranium dust (7) (Figure 5.4).



**Figure 5.4 Uranium refining and conversion**

*(Source: Blind River Facility – Canada)*

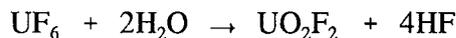
Uranium hexafluoride at room temperature (at which it is stored) is a colourless, crystalline solid with a significant but low vapour pressure. When heated at atmospheric pressure, to facilitate transfer, the crystals sublime without melting and the vapour pressure reaches 760 mmHg at a

temperature of about 56°C. At higher pressures the crystals will melt, and this melting is accompanied by a very substantial increase in specific volume.

The handling of UF<sub>6</sub> in the liquid state represents the greatest potential for high release rates. Experience has shown that more than half of the UF<sub>6</sub> liquid in a container would vapourize in a very short time in the event of a cylinder failure.

Uranium hexafluoride is a reactive substance. It reacts chemically with water, forming soluble reaction products, with most organic compounds and with many metals. Its reactivity with most saturated fluorocarbons is very low. It does not react with oxygen, nitrogen, or dry air.

The main potential hazard following a UF<sub>6</sub> release arises from the reaction between UF<sub>6</sub> and the moisture which is normally present in the atmosphere, producing two toxic substances, hydrofluoric acid (HF) and uranyl fluoride (UO<sub>2</sub>F<sub>2</sub>) according to the equation:



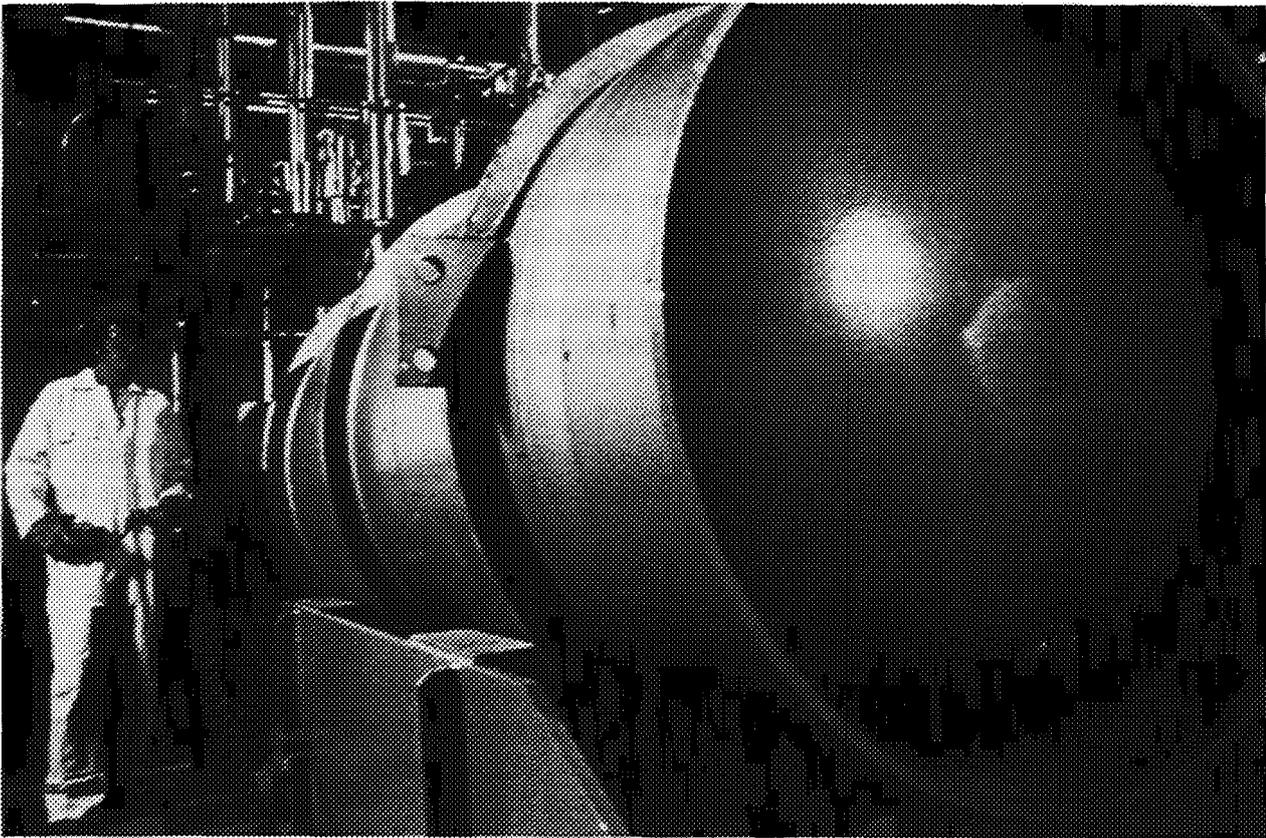
With gaseous UF<sub>6</sub>, this reaction proceeds rapidly. It liberates heat and is accompanied by a substantial volume increase at atmospheric pressure.

Uranium hexafluoride is stored in cylinders and thus the maximum quantity that would be released from any one cylinder is 12 THM, this being the maximum size cylinder in use. If such a release occurred inside a building, the resulting off-site air concentrations of toxic chemicals would probably remain below the dangerous limits (Figure 5.5).

Although the maximum quantity of UF<sub>6</sub> stored in a cylinder is 12 THM, it is possible to have more than one cylinder involved in an accident, i.e. during transportation or at a storage facility. Accidents such as an aircraft crash on a UF<sub>6</sub> production facility are considered to be of very low probability, and are of interest only from an emergency planning point of view; the main risk is a chemical toxicity hazard rather than an immediate radiological hazard.

As far as actual experience is concerned, relatively few accidental releases of substantial quantities of UF<sub>6</sub> have occurred. In 1977 an accident occurred in France (7.1 THM of liquid UF<sub>6</sub> released in about 12 minutes), yet with very limited consequences (7). Studies of the scenarios and consequences of large releases from heated UF<sub>6</sub> containers have revealed that it is principally the employees at the conversion plant who are at risk from such occurrences.

A serious accident involving a UF<sub>6</sub> release occurred in January 1986 when an overfilled UF<sub>6</sub> cylinder ruptured while it was being heated in a steam chest at the Sequoyah Fuels Conversion Facility, Oklahoma, the United States. One worker died because he inhaled hydrogen fluoride fumes, a reaction product of UF<sub>6</sub> and airborne moisture. Several other workers were injured by the fumes, but none seriously. Much of the facility complex and some off-site areas were contaminated with hydrogen fluoride and a second reaction product, uranyl fluoride (8). The results of the off-site impact analysis indicated that no lasting health or ecological harm was done (9).



**Figure 5.5 View of UF<sub>6</sub> cylinder**  
(Source: Port Hope Facility – Canada)

Some countries have chosen to reduce the risks associated with the storage of UF<sub>6</sub> by converting depleted UF<sub>6</sub> into UF<sub>4</sub>, a powder which can be stored more economically and with less risk of serious accidental release. The safety concerns with this conversion activity, to both the workers and the public, are the same as those associated with the initial production and handling of natural UF<sub>6</sub>. The risks associated with this conversion process are balanced against the risks associated with the storage of UF<sub>6</sub> to justify whichever option is chosen based on the ALARA principle.

### **5.1.3 Uranium enrichment**

Beginning in the early 1940s, the main route for uranium enrichment was through gaseous diffusion processes. Only in the late 1960s did centrifuge plants begin to take over some of the enrichment work, and since then there has been significant decommissioning of diffusion plants (see section 5.3.3). Thus a considerable amount of experience has been amassed from construction, operation, refurbishment and decommissioning of diffusion plants.

#### **5.1.3.1 Diffusion processes**

These plants invariably process uranium hexafluoride. The corrosive and toxic hazards associated with UF<sub>6</sub> are covered generally in 5.1.3.6. Nuclear/radiation hazards are also covered generically in 5.1.3.5 and 5.1.3.6.

Diffusion plants are designed for economic reasons to be run continuously for periods of several years at one time. Reliability of process equipment and components is high. From a safety standpoint, too, maintenance should be minimised in all areas of the plant where  $UF_6$  is present and particularly in high-enrichment areas where radiation doses could be significant. Bearing in mind that a diffusion plant comprises both pressure systems and vacuum systems, driven by massive pumps, it is not surprising that the most common problems are associated with vibrations leading to pump and valve leakages and even failure (10). Vibration could lead to blade failure, and missiles could sever  $UF_6$  lines and lead to a release into the atmosphere. Such a release is by far the greatest threat to personnel and public from enrichment plants, and is covered in 5.1.3.5. With many years of operational experience to draw on, such problems are now much less frequent. Modularisation of plant can ease on-line maintenance.

Operational experience with diffusion plants has been good. The facility operations have generally been reliable and have performed well with low levels of  $UF_6$  release during normal operations.

#### 5.1.3.2 *Centrifuge processes*

After its development in the 1960s and 1970s, uranium enrichment by gas centrifuge is a proven, commercially viable, technology. Gas centrifuge enrichment plants pose hazards associated with handling  $UF_6$  (5.1.3.5) and enriched uranium (5.1.3.6), as well as hazards that are also found in other industries – particularly those posed by the use of high-speed rotating equipment. The plants are designed to be run continuously for periods in excess of ten years, and reliability and ease of maintenance are of paramount importance both for safety and economic reasons (see, for example [14], and Figure 5.6).

Separative efficiency is greatly enhanced by increasing the peripheral speed of the centrifuge rotor. The speed is limited by *i*) the tensile strength of the material of construction and *ii*) resonant vibration. The latter may be reduced by using short rotors and heavy-duty bearings of minimal friction (11). Typical peripheral speeds for aluminum rotors are 350 meters per second.

A wreck of a gas centrifuge unit operating at these high speeds would generate shrapnel as a result of the destruction of the rotor and other spinning components. The probability of such a wreck is minimised by good design and construction (12).

Research into new constructional materials continues (13) in order to develop safer and higher-performance centrifuges. Material stresses are at the heart of the safety of centrifuge plants.

Operational experience with centrifuge plants has been good; plants have generally been very reliable and performed well with low levels of  $UF_6$  release during normal operations, and there have been very few and only minor mechanical failures.

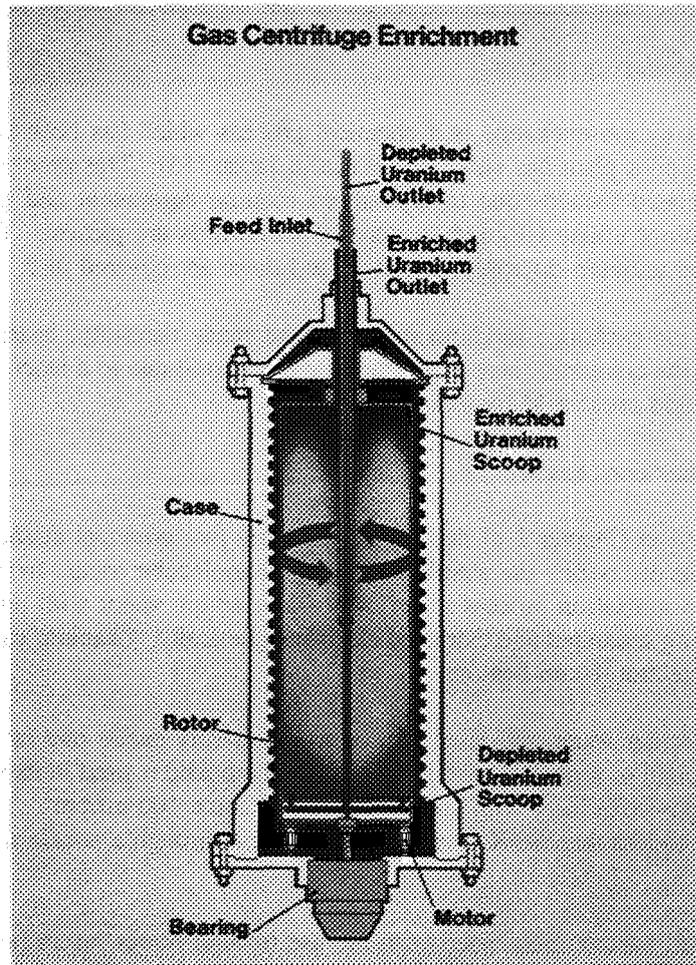
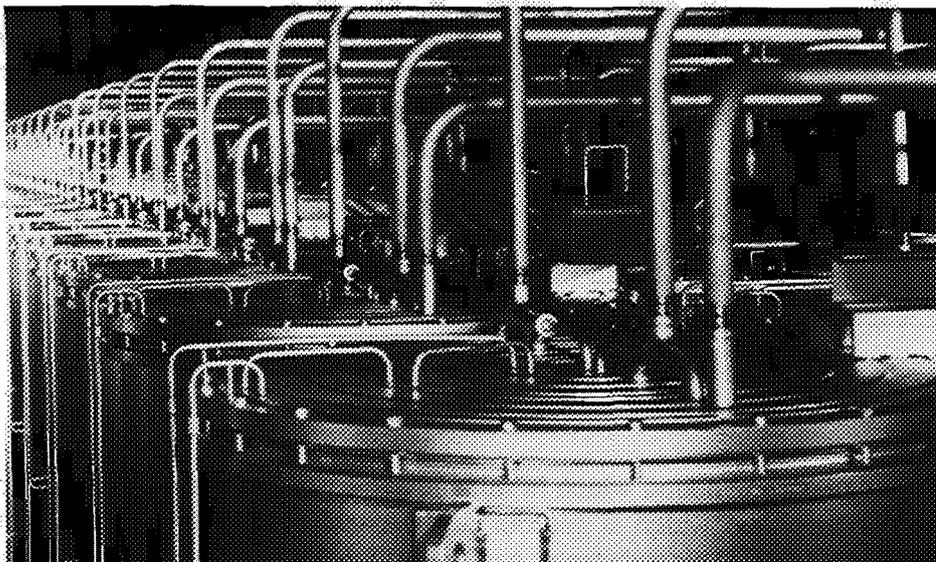


Figure 5.6 Gas centrifuge enrichment  
 (Source: PNC – Japan)



Ningyotoge centrifuge enrichment

### 5.1.3.3 Chemical processes

Enrichment by ion-exchange is a chemical process which has been studied in recent years (11), and a pilot plant has successfully run in Japan (13). This process does not have any of the associated hazards due to high-speed rotating equipment and pressure/vacuum systems. It does not utilise  $UF_6$  and so none of its hazards are present. The problems of handling enriched uranium do apply, however, and these are covered in 5.1.3.6.

Choice of ion-exchange medium must also be made carefully, bearing in mind the reagents used in the columns, since explosions have occurred due to a reaction between ion-exchange resins and nitric acid for example.

### 5.1.3.4 Laser enrichment

As described in Chapter 2 of this report, there are two principal methods of enriching uranium by means of lasers – the atomic route and the molecular route. In the atomic route, a multistep photo-ionization process is performed in uranium vapor with tunable lasers, the resultant ions being separated and extracted electromagnetically. Uranium vapor, at  $2,000^\circ C$ , is produced by electron-beam heating.

In the molecular route, the laser is used to selectively excite one isotopic form of a molecular compound of uranium, such as  $UF_6$ . The compound is used in its vapour state and requires a very powerful laser with an energy flux of some  $100 \text{ mJ/cm}^2$ . The gaseous  $UF_6$  has to be cooled to less than 80 K, usually by expansion through a nozzle when mixed with nitrogen.

Conventional hazards associated with the use of high-power lasers (i.e. high voltages and high-power laser beams) are common to both methods. In addition the possible toxicity of laser dyes and potential fire/explosion hazards associated with high-pressure dye/solvent circuits are taken into account during safety assessments. Containment of high-temperature corrosive uranium vapour poses some metallurgical problems. The vessel will have to be designed to withstand/prevent any overpressurisation resulting from contact of liquid uranium with cooling water. An additional problem with the atomic route is the removal of the uranium-235 metal from the collector plates. Hazards associated with  $UF_6$  are described in 5.1.3.5, but the combination of the corrosive nature of  $UF_6$  and thermo-mechanical constraints make it difficult to find suitable materials for the molecular route.

Despite the many difficulties, it has been reported (11) that in the United States, substantial progress has been made in developing a large-scale laser enrichment process. In Japan two laser isotope separation processes are under development (13) and in France an atomic-vapour laser process, AVLIS (SILVA, in French), is being developed (15).

### 5.1.3.5 Uranium hexafluoride hazards

The only uranium compound which is suitable for diffusion, centrifuge or laser enrichment (by the molecular route), and which is gaseous at reasonably low temperatures ( $40$  to  $60^\circ C$ ), is uranium hexafluoride,  $UF_6$ . This substance has the additional advantage that fluorine is mono-isotopic. However  $UF_6$ , or "hexafluoride", is a powerful fluorinating agent. It reacts with almost all metals to form metal fluorides. It reacts, often explosively, with organic material to form fluorinated compounds and hydrogen fluoride. With water or almost any level of water vapour, it forms highly

corrosive and toxic hydrogen fluoride gas, together with a number of solid and toxic compounds of uranium, oxygen, fluorine and water of crystallisation.

Containment of  $UF_6$  is of prime concern, from the point of view of safety, at all stages of the enrichment process. Typical containment materials, such as stainless steel and copper, retain their long-term integrity by virtue of the immediate formation of a passive, impervious, protective fluoride layer when first contacted by hexafluoride which prevents further corrosion and thinning of the containment.

Processing facilities are designed to a high degree of leak-tightness, particularly where hexafluoride is under reduced pressure, in order to prevent the ingress of moisture and the formation of solid  $UO_2$ ,  $F_2$  and related compounds which could build up and cause blockages.

It is those two requirements, i.e. to contain  $UF_6$  within a plant and to keep air and moisture out, which tend to dominate most of the tasks performed in enrichment plants.

$UF_6$  is supplied to the plant in large, typically 12 THM, containers. At room temperatures it is solid, and the containers are warmed to generate  $UF_6$  as a gas. Overfilling  $UF_6$  storage containers can lead to rupture, due to expansion of  $UF_6$  solid on melting (accompanied by a significant volume increase). This has resulted in at least one incident at enrichment plants (8, 9). At the end of the process the depleted hexafluoride, or "tails", is recondensed into containers which may be stored outside. Having little commercial value, stocks of depleted  $UF_6$  have increased dramatically over the years, and can present a hazardous inventory to local populations in the event of a failure of containment accompanied by a source of heat, e.g. an aircraft crash. Storage of tails as depleted uranium oxide or uranium tetrafluoride would appear to be a more acceptable longer term alternative, although there are hazards associated with the conversion (see 5.1.2).

Overall the primary hazard associated with enrichment plants is the toxicological effects of  $UF_6$  should it get into the environment. Therefore, of paramount importance in any plant is a comprehensive leak detection and alarm system. For example the Eurodif plant at Tricastin, France, (10) and the BNFL plant at Capenhurst, the United Kingdom, and JNFL, PNC plants in Japan have systems for detecting and containing accidental  $UF_6$  leaks.

#### 5.1.3.6 *Nuclear/radiation hazards*

##### *i) Criticality*

Nuclear criticality is always a concern when working with enriched uranium. Criticality is not a major concern in centrifuges themselves because the nuclear inventory in the machines and process piping is insufficient to form a critical mass under all credible scenarios. Criticality is controlled in other areas of the centrifuge plant, and in other enrichment processes, through the use of "geometrically safe" equipment, control of moderators (or a combination of the two), as well as adherence to administrative control of the process. In view of the size of equipment used in diffusion plants, criticality remains a potential problem. In other enrichment processes, the criticality problems can generally be resolved by engineered means. Criticality assessments, based on an assumed maximum level of enrichment, could be invalidated by an accidental over-enrichment in the process and HAZOPS type of techniques could highlight areas where this is a potential problem.

## ii) Recycled uranium

Until recently most enrichment was carried out only on uranium supplies which had not been irradiated in a reactor. However, considerable progress has been made within existing nuclear fuel cycles to make use of recycled uranium (19). In particular URENCO has undertaken comprehensive tests and trials with recycled uranium (16), and by 1986 some 125 THM of such uranium had been re-enriched.

Recycled uranium differs significantly from natural uranium in three certain respects:

- Uranium-232, which does not occur naturally, is present in recycled uranium. It has a greater specific activity than uranium-235, and its decay chain includes a 1.9 year half-life thorium-228 alpha-emitter and a number of hard gamma emitters, notably thallium-208. The gamma activity increases with time reaching a secular equilibrium, much greater than natural uranium, in about ten years.
- Uranium-234, a naturally occurring alpha emitter, is also enriched in fuel before irradiation. It is only partly burnt-out during irradiation. As a result, recycled uranium has a higher alpha activity than natural uranium. Traces of plutonium and neptunium will also contribute to alpha activity of recycled uranium.
- Traces of ruthenium-106 may contribute to a higher gamma background (19).

Thus to the safety problems already covered there is added one of radiation dose levels when recycled uranium is enriched again. The presence of uranium-232 should encourage utilities to recycle material quickly, but economic factors up to now have resulted in little recycling of uranium. Most of the present stockpiles (estimated at 29,000 THM by 2001 in the United Kingdom, France and Germany (16)) of  $UO_3$  or uranyl nitrate, from reprocessing, are already at or near equilibrium gamma dose levels.

### 5.1.4 Fuel fabrication

For more than two decades the fabrication of uranium fuel elements for GCR and Magnox-reactors, AGR, LWR and HWR has been an industrial enterprise (20, 21). More than 5,000 THM uranium oxide fuel elements are manufactured each year for LWR in OECD countries. Also the fabrication of plutonium-containing mixed oxide fuel elements for FBR and thermal reactors has reached industrial maturity on a modest scale of throughput. Very few accidents involving significant radiological hazards and substantial release of radioactive material to the environment have been reported from the nuclear fuel fabrication industry. In addition, no criticality accident has occurred. The reasons for this very good safety record are the well-defined processing and handling operations and associated safety procedures, and the chemically stable solid state of most of the radioactive materials involved. The main exceptions to the latter are  $UF_6$  conversion and wet scrap recovery phases which require liquid processing. Potential  $UF_6$  release is also of concern in fuel fabrication plants.

Mixed oxide fuels are used for both fast reactors and thermal reactors. Due to the fact that the implementation of FBR in many countries has been postponed or even abandoned, a growing interest in recycling plutonium in LWRs can be observed in some countries. Since most of the plutonium available for thermal recycle in the future will be recovered from LWR spent fuel with increasing burn-up, the design of new planned MOX fuel fabrication facilities has to be based on

typical LWR-plutonium isotopic compositions and PuO<sub>2</sub> contents of 3 to 8 per cent in the MOX-fuel for LWR.

For the conversion of UF<sub>6</sub> to uranium metal or UO<sub>2</sub>, wet and dry conversion processes are used on a large industrial scale. The subsequent processes of powder production, pellet pressing and sintering, rod fabrication and assembling of fuel bundles are fully developed. Completed fuel assemblies are stored at the fabrication plant before being transported to the reactor. This storage is regarded as part of the fabrication plant.

The conversion of plutonium nitrate solution to PuO<sub>2</sub> is part of the recovery of plutonium at the reprocessing plants in most cases. Some MOX fuel fabrication plants, however, are also equipped with facilities for plutonium nitrate conversion and for dissolution and purification of plutonium-containing scrap material. Several chemical conversion processes for plutonium nitrate solution or co-conversion processes for uranium-plutonium solutions are industrial standard. The following steps of MOX-powder and pellet fabrication and the manufacturing of MOX fuel assemblies for FBR and LWR are technically very similar to the fabrication of UO<sub>2</sub> fuel assemblies.

#### 5.1.4.1 *Uranium fuel fabrication*

Due to the low radiotoxicity of slightly enriched uranium, only limited off-site environmental consequences are to be expected following accidents. However, as in the case of the enrichment process, the safety significance of accidental releases may require a more careful assessment when recycled uranium arising from spent fuel reprocessing comes to be used on a larger scale. In general the special and inherent design safety features and carefully planned operational procedures reduce, to very low levels, the probability of significant accidents due to plant malfunctions. Both these and external events are considered here.

In the design phase of fuel fabrication plants internal safety hazards such as fire, explosion, release of UF<sub>6</sub> and criticality have to be considered. Adequate precautions are incorporated in the design to reduce the probabilities of these accidents to an acceptable level. Of equal importance is the safe and reliable operation of the facility because operational experience clearly indicates that non-routine operations and human error are main contributors to incidents.

Various kinds of external events could have significant consequences for both the plant itself and the environment, and these too have to be considered in the safety assessment. Flooding the facility could create the risk of a criticality accident, therefore the plants are usually built in a very low flood risk area. However, if the absence of flooding cannot be ensured design precautions should be taken to prevent the plant from becoming critical in the flooded state. The main hazard in case of an external event such as earthquake or aeroplane crash arises from the possible release of uranium hexafluoride.

Specific safety studies have been performed and safety criteria have been established, in addition to the safety analysis, within the licensing procedure for fuel fabrication plants in some countries (22, 23, 24, 25).

The following possibilities for in-plant-accidents are considered to be relevant:

- release of UF<sub>6</sub> caused by a pipe or valve failure or rupture of a heated UF<sub>6</sub>-container;
- criticality;
- explosion in a fluidised bed or sintering furnace.

Spills or leakages of uranium containing solutions, release of uranium dust due to filter failures, and fire in the scrap purification are other possible incidents. The main risks, albeit small, to personnel during normal plant operations are associated with *i*) the potential inhalation of uranium oxide fine particles when working with powders and *ii*) exposure to external radiation during fuel inspection and storage.

Releases of  $UF_6$  caused by failures of pipe connections or valves have occasionally occurred in fuel fabrication plants. Minor or major contamination with  $UO_2F_2$ , within the conversion building, from the reaction of the released  $UF_6$  with moisture, was the main result of these incidents. No off-site consequences occurred. To prevent the need for major clean-up work after a  $UF_6$ -release, a separation or enclosure of the  $UF_6$ -conversion plant area is recommended. Rupture of a heated  $UF_6$ -cylinder in the evaporation station at the head end of the conversion process, represents the most severe possibility for a large  $UF_6$  release. No incident of this kind and severity has ever occurred in a fuel fabrication plant. This type of accident has been considered earlier in section 5.1.3, and also in section 7.3 on accident experience. In the case of enriched uranium the capacity of  $UF_6$ -cylinders of type 30B is 1.5 THM uranium. Studies indicate the maximum release of  $UF_6$  in case of a rupture of a heated cylinder for enriched uranium would not be in excess of approximately half of this quantity. Rupture of the conversion furnace itself would only give rise to a small leakage because of the restricted inflow rate used. Modern evaporation stations for  $UF_6$  are equipped and operated with remotely controlled or automatic valves to keep as low as possible the quantities of  $UF_6$  released in case of an incident.

One of the major safety concerns in a commercial fuel fabrication plant and the associated fuel store is the potential for a criticality. Since in most facilities only low-enriched uranium is handled (with enrichments up to 5 per cent uranium-235), fabrication steps which are made under dry conditions are inherently safe in the absence of a moderator. In the steps in which hydrogenated fluids are used, mass or geometry controls are applied. In some cases fixed neutron absorbers are additionally used to maintain subcriticality even in the event of accidental moderation and neutron interaction. Off-site exposures from any fission products generated in criticality accidents are likely to be insignificant. The main hazard will be the local high-radiation fields which are produced by the nuclear excursion.

The increased use of recycled uranium in the future will not pose any new or increased criticality risk. Also the increase of burn-up probably requires only a small increase of the initial enrichment so that fuels for LWR with more than 5 per cent uranium-235 are unlikely to be fabricated in large quantities in the foreseeable future.

Criticality excursion characteristics for moisture-containing under-moderated powders should be further investigated (30, 31). Also a continuation of criticality excursion experiments for solutions similar to the CRAC series (32) could support a better understanding of critical excursions. Release fractions of all significant fission products should be measured as part of this. Any further increasing of uranium mass limits in fuel fabrication stages, which would take into account a nearly zero moderation, should be based on the results of specific experimental criticality programmes on low-moderated and low-enriched uranium.

The use of reprocessed uranium, in fuel manufacturing, with higher concentrations of uranium-232, uranium-234 and uranium-236 induces a higher gamma radiation field due primarily to the formation of the decay products bismuth-212 and thallium-208. With increasing burn-up the concentration of ruthenium-106 will also increase, but this effect could disappear after a prolonged storage. The radiological impact of uranium-232 and ruthenium-106 are both time-dependent but in

an opposite way. If large quantities of reprocessed uranium are to be handled, improved shielding and encapsulation of uranium are desirable. The level of alpha or gamma-emitting impurities in reprocessed uranium will call for improved purification of the  $UF_6$  at the end of the conversion steps and for a substantially improved cleaning of liquid effluents. Studies of these effects and their possible influence on accidental situations should be performed taking into account not only fuel manufacturing but the complete recycling strategy for reprocessed uranium.

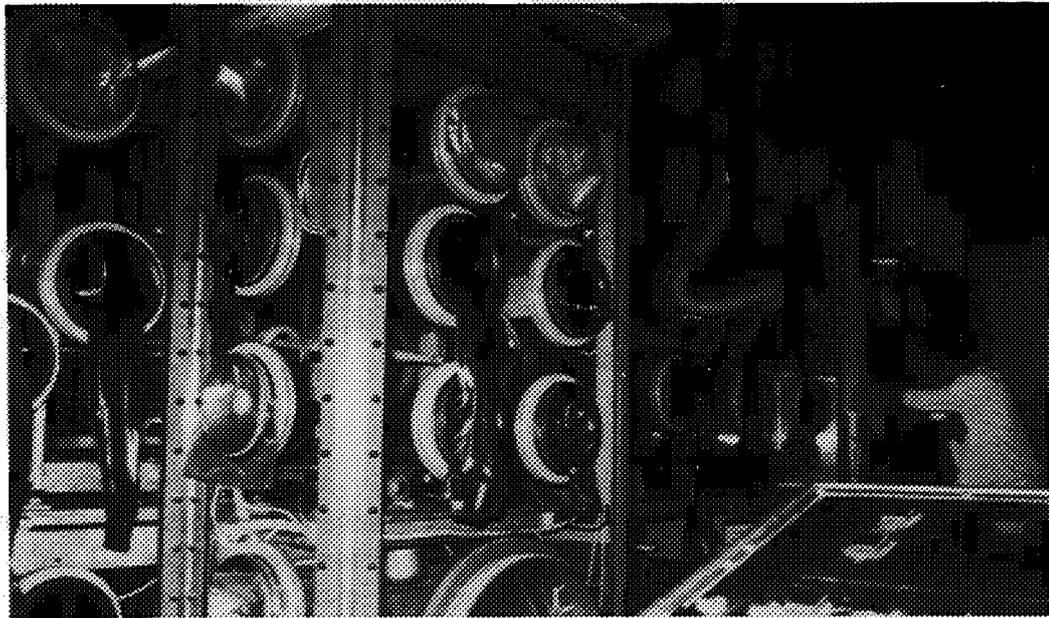
In fluidised bed or sintering furnaces, hydrogen is used as a reducing agent, usually diluted with inert gas to prevent the risk of explosion. However explosive mixtures can develop from incomplete removal of air from a cold furnace at start-up, or from in-leakage of air into the furnace. Control mechanisms have a low probability of failure, but the possibility of the development of an explosive air-hydrogen mixture cannot be completely excluded. The pressures developed would be sufficient to blow out large quantities of  $UO_2$  powder, if an explosion occurred in a fluidised bed furnace. For a sintering furnace explosion,  $UO_2$  would be blown out of the ends of the furnace in the form of pellets. The release of  $UO_2$  to the environment would, however, be small and its off-site consequences would not be of any significance.

#### 5.1.4.2 *Mixed oxide fuel fabrication*

Historically, MOX fuel technology was developed in several countries, namely, Belgium, France, Germany, Japan and the United Kingdom (for descriptions see sections 2.2 and 4.2 and, Figure 5.7). Due to the high radiotoxicity of plutonium, the safety provisions and safety-related design characteristics of a mixed oxide fuel fabrication facility are quite different compared to those for uranium metal fuel manufacture (168). To prevent the dominant risk of plutonium release from a MOX fuel cycle facility, a multiple barrier system, supported by a system of graduated low pressures is designed to contain the radioactive material. The design of the technical equipment has to be compatible with this enclosure which usually is provided by glove boxes, tight metallic enclosures and caissons. In addition, the emission of neutrons and gamma radiation from plutonium and the associated americium requires adequate shielding. In the design of new plants planned for larger throughputs, and for LWR-plutonium with relatively high content of plutonium-238 and americium-241, remote or automated operation of most of the equipment is usually provided to reduce radiation exposure to workers. A computerised MOX fuel fabrication has been introduced in PNC Tokai Plant (Figure 5.8). Since the containment of plutonium and the reliability of the multiple barrier system are of paramount importance, even quite low-probability initiating events with the possibility of damaging the system of barriers are to be considered as possible causes for incidents during the assessment process. In the safety-related design, adequate protection has to be provided especially against fire and explosion risks.

External events also have to be considered in the safety assessment of a mixed oxide fuel fabrication plant. Possible consequences such as a criticality accident or wide-spread contamination may arise from flooding and require close study, if site characteristics or design provisions do not rule out such an event. Other external events such as explosions from chemicals outside the facility, fires in adjacent industrial plants or forests, earthquakes and aeroplane crashes may pose a significant risk of release of plutonium bearing material. A thorough analysis of these risks has to be performed taking into account site characteristics, accident probabilities and the performance of the containment system. As a result of these studies strong concrete buildings have, in some cases, been planned or built to protect the facility against any damage from earthquake or tornadoes, pressure waves from explosions and, if necessary, penetrating missiles from aircraft crashes.

Specific safety studies and safety criteria have been devoted to mixed oxide fuel fabrication (22, 23, 24, 25). In a mixed oxide fuel fabrication plant, among all possible internal accidental events, fire and criticality are identified by accident analysis as those which outweigh all other internal events with respect to their consequences.



**Figure 5.7 Fuel fabrication plant**  
(Source: Dessel – Belgium)

Fire and criticality events may give rise to a significant hazard which may have off-site environmental effects, even if small quantities of plutonium dioxide penetrate the plant filter system which includes multiple high-efficiency particulate air (HEPA) filters. Adequate protection of these filters from pressure waves, moisture and fire, therefore, is vital. As in the case of uranium oxide plants, criticality accidents can only lead to off-site exposures if iodine and noble gases are released as a result of the nuclear excursion. Even though there is no retention of these isotopes, such releases are likely to be insignificant.

Evaluation of the amount and form of radioactive material which can be released from a plant under accident conditions forms an important part of safety assessments, and continued effort has been devoted to both related analytical and experimental work in this area. Work of this nature is relevant to many potential fuel cycle accident studies, but is particularly important for plutonium handling facilities which may release plutonium aerosols in the event of fires. The development of analytical tools and computer programmes to simulate the release of plutonium aerosols, their deposition, and the behaviour of ventilation and filtration systems should be continued further.

The plant criticality safety design principles which are required are similar to those discussed above for the uranium oxide fuel fabrication plants. In this case, however, mass control is necessary even in the dry-processing stage since, for certain combinations of mass and geometry, criticality can be achieved with unmoderated plutonium dioxide. A close examination of storage facilities for large stocks of plutonium or mixed oxide has to take into account not only for criticality risks, but also shielding requirements and adequate removal of decay heat. For process steps with dry fissile material the restriction and control of moderation is essential. Specific criticality safety precautions have to be taken at the MOX powder mixing stage.



**Figure 5.8 MOX fuel fabrication**

*(Source: Tokai – Japan)*

Unlike a criticality excursion or an explosion, a fire is usually not a very rapid event and often starts from a small flame source. Those responsible for the design, construction, and operation of fuel fabrication plants consider in detail the possibility of fires, and provide the equipment and procedures necessary for their prevention. Some national recommendations indicate methods for the establishment of fire protection programmes which should prevent, detect, extinguish, limit or control both fires and explosions and their accompanying hazards and damaging effects (25, 26, 27). Design precautions include the partition of the plant into "fire and contamination sectors". If these precautions are taken, the probability of having a fire of serious magnitude is considered to be very low. In general, operators have time to react and extinguish any small fires which occur.

The process materials, oxides of uranium and plutonium, are not themselves flammable but some items of equipment, such as glove box panels, may be made of combustible materials. In addition a fire risk might be present from stored combustible waste, combustible fluids and possible leaks of flammable gas.

The main plants' HEPA filters are normally located some distance from such equipment. This remoteness, together with the provision of a valve and, in some cases, an extract duct cooling arrangement located some distance upstream of these filters, are usually sufficient to protect them against the effects of fire. In safety assessments the fire is, however, assumed to destroy the local filters located on the glove boxes so that any plutonium or uranium oxides which become airborne reach the final main plant filters. Therefore the latter, which include at least two HEPA filters, should be designed to retain these oxides. Blockage of filters in fire situations is a condition which

requires consideration. Fires involving glove boxes can give rise to substantial local release of radioactivity within the plant and this can lead to occupational hazards.

Plant-specific studies are recommended on the complex protection system in case of a fire and the requirements for on-going ventilation, smoke removal and rescue of personnel. Also the effect of various fire-fighting agents on the plant and the safety of the ventilation system are of specific interest. The design of new facilities should assess the benefit of fire-extinguishing systems. In evaluating the benefits of these systems' secondary effects like criticality risks, over-pressurisation of rooms and filters, deposition of combustion products on filters and protection of personnel from toxic gases have to be considered. Computer programmes have been developed to simulate the generation and propagation of fires in glove boxes, rooms and ventilation systems providing tools for an in-depth analysis of these accident scenarios (28, 29).

In a mixed oxide fuel fabrication plant, explosions are conceivable in the preparation of powder in the conversion furnace, in pellet production at the sintering stage, and from radiolytic hydrogen generation if large amounts of plutonium are stored as liquids. Generally the explosion risk is low as long as adequate provisions against accumulation of hydrogen are taken. In furnaces, hydrogen is generally diluted with an inert gas. Redundancy is provided in the gas-supply control system to prevent gases with high concentrations of hydrogen from entering the furnaces. Hydrogen monitors are provided in some glove boxes or rooms. The consequences of an explosion are generally similar to those of a fire.

In the case where dirty scrap recovery is performed, additional fire and explosion risks have to be taken into account, e.g. solvent explosion in a glove box, explosive decomposition of ion exchangers, and hydrogen explosion. Similar risks are present in wet americium separation processes. Utmost care is taken to design the equipment and its peripheral safety features and considerable attention must be given to the location of the recovery facility within the MOX fuel manufacturing site.

Other local plant incidents that can occur in a mixed oxide fuel fabrication plant include loss of an alpha-tight window, glove failure or leakages of plutonium-containing liquids. These incidents can result in contamination within the working rooms, but off-site effects are practically zero. Very limited and minor off-site contamination may be expected from failures of ventilation systems and filters. Because normally several filters are arranged in series, a single filter failure will not result in any contamination of the environment. Nevertheless adequate design, protection and in-service control of filters are all important to prevent any release of plutonium.

Design provisions for *i)* relatively easy decontamination, maintenance and repair, *ii)* management of abnormal events, and *iii)* decommissioning of plant equipment should continue to be investigated. Also the prevention of incidents during decommissioning work should be part of these studies.

The future recycling of plutonium from advanced fuels, which will have higher concentrations of plutonium-236, plutonium-238, plutonium-241 and americium-241 or residues of fission products, will create the need for better protection against radiation, especially from neutrons, and will increase the heat production (168). Remote operation, automatic process operation and control by process computers may help to reduce the occupational radiation exposure. On the other hand, it has to be carefully investigated if failures of automatic devices or computer hardware or software have the potential to generate new incidents.

Safety problems related to a prolonged storage of plutonium in various forms, and of plutonium-containing scrap material, should continue to be kept under review. Degradation of plastic material by decay heat and radiation, corrosion and increasing build-up of americium-241 may lead to problems in handling and further use of these materials. Dilution of aged plutonium batches with fresh plutonium is one possible solution. Special chemical purification to reduce the americium-241 content is another solution.

Criticality experiments with low moderated mixed oxide and experimental studies on the behaviour of heterogeneous fissile systems like moderated pellets or fuel rods in case of a criticality accident are considered to be desirable.

## **5.2 Back end of the fuel cycle**

The back end of the fuel cycle starts with the unloading of spent fuel from a nuclear power plant and incorporates all operations leading to recoverable fissile materials, wastes and spent fuel if declared waste. This part of the report covers the following stages:

- 5.2.1 Spent fuel storage
- 5.2.2 Fuel reprocessing
- 5.2.3 Radioactive waste management
- 5.2.4 On-site interim storage of wastes

### **5.2.1 Spent fuel storage**

Several technologies are being used for the storage of spent fuel at reactor (AR) sites and at sites away from reactors (AFR) (Figure 5.9). Both wet (pool) storage facilities and dry storage facilities (buildings and containers) are used on a commercial scale.

The safety of spent fuel storage has been extensively evaluated. The US Nuclear Regulatory Commission (NRC) reported in the "Waste Confidence Decision" of 1984 (33) that there is reasonable assurance that spent fuel can be stored safely and without significant environmental impact in reactor pools or in AFR spent fuel storage installations. For both dry storage and wet (pool) storage, the NRC stated its belief that current storage technologies are capable of providing safe storage for at least 30 years beyond the active lifetime of the reactor facility. The NRC also concluded that the possibility of a major accident or sabotage at a spent fuel storage facility with radiological consequences for the public is extremely remote. The BEFAST I report (34) and the recent survey of experience in the storage of nuclear fuel by the IAEA (35) both support the confidence in longer term wet and dry storage. In the IAEA report on fuel cycle status and trends (36) it is stated that there is a general consensus that feasible and proven technologies have already been developed and exist for short- and long-term storage of spent fuel.



**Figure 5.9 Pool storage**

*(Source: BNFL's Sellafield Plant – The United Kingdom)*

Considerable experience has been gained in the transport of spent fuel elements and in the consequent safety-related development of suitable transportation casks. This experience has made it possible to develop a concept for dry storage of spent fuel elements within transportation casks; dry storage containers generally have not been the transportation casks themselves. At Gorleben in Germany, an AFR dry storage facility that uses this concept has been licensed. This concept is based on three essential safety requirements:

- The casks for dry storage of spent fuel elements must satisfy the stringent requirements of type B(U) transport casks (38, 39, 40) for the whole lifetime of the facility.
- A system of quality assurance and inspection is instituted to guarantee that the requirements for type B(U) casks identified above are satisfied.
- The cask has to fulfil the "two barrier principle", which means that the release of radioactive materials into the environment has to be prevented by a double-lid system together with the equivalent very strong cask itself.

The concept of the so-called "dual purpose" (Storage/Transport) cask has recently (41) been licensed in the United States in the framework of a new policy of dry storage in Independent Spent Fuel Storage Installations (ISFSI). According to this new policy the reactor operators are entitled

to store the spent fuel elements, which have cooled in a pool for at least one year after discharge from the reactor, in specially licensed containers under dry conditions for 20 years or more. A number of storage casks have received official approval for that purpose, e.g. CASTOR V21, MC10, NAC S/T, NAC-C28 S/T, etc.

The US DOE is sponsoring a study to determine the effects of a long-term normal storage (20 to 30 years) on the performance of a storage cask in transportation. Included in the study are the effects of radiation, corrosion and heat on the container. The safety problems associated with the short-term storage of spent fuels arise from the high-fission product content of the fuels and the presence of fissile materials. The beta-gamma activity in LWR spent fuel six months after discharge from reactors still amounts to about 150 TBq per THM. The plutonium produced in a PWR fuel assembly is about 4 kg. Thus, a 1,000 tonne uranium fuel storage facility will have a plutonium inventory on the order of 9 THM and the maximum heat evolved from the fission products in the fuel after one year's cooling time is about 12 MW.

Although the inventory of radioactivity is high, dispersion and release of any significant proportion of it cannot take place either rapidly or readily.

#### *5.2.1.1 Wet storage facilities*

##### *At reactor (AR) pool storage*

This is a highly developed technique which is the standard method used worldwide for short-term storage of spent fuel. The continuous production of spent fuel by the power reactors all over the world and the delay in developing the required reprocessing capacity or long-term storage facilities on geological repositories urged the reactor operators to increase the pool storage capacity by enlarging existing pools, building new facilities and increasing the specific fuel density in the pools (re-racking).

##### *Away from reactor (AFR) pool storage*

AFR wet storage facilities have been commissioned and are now in operation, notably the CLAB facility at Oskarshamn in Sweden and the TVO-KPA facilities at Olkiluoto in Finland. The CLAB facility was commissioned in 1985 and has a capacity for 5,000 tonnes of uranium. At present about 1,600 tonnes uranium are in storage and CLAB is expected to reach full capacity by the year 2004. The storage section of the facility has been constructed entirely under ground in a rock formation. The TVO-KPA facilities began operation in 1987 and have a capacity for 1,200 tonnes uranium.

##### *Pool storage capacity at reprocessing sites*

In France, Cogema has been engaged in an extensive programme to expand the wet storage facilities at La Hague. The total capacity of the facilities amounts to 11,200 tonnes uranium. Seismic considerations have been taken into account for these new pools where the pool structures are supported on visco-elastic pads.

At Sellafield in the United Kingdom, the storage capacity has been expanded with an expected total capacity of over 7,500 tonnes uranium. At Tokai in Japan the storage capacity is now 140 tonnes uranium, and at Rokkasho the storage capacity will be 3,000 tonnes uranium.

*Safety issues with wet storage of spent fuel*

Table 5.2.1 compares fuel conditions in the reactor and in the storage pools, emphasizing the much more benign environment of the latter. Temperatures, pressures and radiation fluxes are lower in the pool, and there is no intrinsic driving force for the sudden release of a major fraction of the radioactive materials contained in the stored spent fuel even under abnormal operating conditions.

Table 5.2.1 Comparison of conditions for water reactor fuel (42)

	In-reactor	In-pool
Fuel temperature (°C) (peak centerline)	1,200 - 1,700	100
Water temperature (°C)		
BWR	270 - 300	20 - 50
PWR	320 - 340	20 - 50
Clad (inside) surface temperature (°C)	340 - 400	30 - 60*
Gas pressure MPa**		
BWR	0.5 - 5	0.2 - 2
PWR	4 - 5	1.7 - 8
Fission gas evolution (calculated in per cent)		
BWR	2	negligible
PWR	15	"
Surface heat fluxes (W/cm <sup>2</sup> )	up to 80	0.03***
Radiation fluxes (max)		
Neutron (n/cm <sup>2</sup> sec) > 1 MeV	$3-6 \times 10^{13}$	$10^5$
Gamma (R/h)	$10^9$	$10^5$
<p>* After cooling for several weeks; the exterior surface temperature is ~ 10°C above the bulk water temperature; the interior and exterior clad surface temperatures are essentially the same at pool storage conditions.</p> <p>** 1 MPa 10 bar = 145.04 psi. Values quoted in original reference have been rounded off.</p> <p>*** For intact fuel rods.</p>		

The zircaloy or stainless steel cladding of the fuel rods is the most efficient barrier against fission product release during handling and storage of spent fuel bundles. Consequently, knowledge of those cladding properties which determine the fuel rod integrity is important. Visual examination and radiation monitoring have revealed no evidence of any degradation in the stainless steel or zircaloy clad uranium oxide fuel which has been stored for periods of 10 to 20 years. Recent examination of some CANDU fuel stored for 27 years revealed no deterioration of the cladding (159). The situation is less favourable with Magnox clad metallic uranium fuel, but only if the cooling water chemistry is incorrect when there then may arise some indications of the occurrence of progressive cladding failure. Mechanical damage during reactor discharge and fuel handling in the pools appears to have been minimal.

The hydrogen concentration found in zircaloy fuel rod cladding at the end of reactor residence time is typically about 100 parts per million. When the fuel pin is cooled to approximately 40°C, after shutdown, a large part of the hydrogen in the cladding tube is precipitated in the form of finely dispersed hydrides. It is known that this precipitated hydride enhances the cladding strength although ductility is reduced. Since no fuel deformation occurs during handling operations, this reduction in ductility should not result in any additional defective rods entering the storage pool. Experience endorses this assertion.

It is known that for longer storage times at normal pool temperatures of approximately 40°C no increase in the size of hydride precipitation occurs.

There are slight temperature differences between the extremities of the stored fuel assemblies, caused by the upward coolant flow, but no redistribution of hydrogen by thermal diffusion mechanisms is anticipated because of the very low diffusion coefficients and the negligible solubility at this low temperature.

Given adequate control of water purity, which is important in the corrosion context, it is known that zircaloy is virtually immune to corrosion by oxidation in water under the long-term storage conditions of fuel assemblies. Likewise, fission product induced corrosion will be negligible.

Cladding stress corrosion cracking phenomena have been considered. Caesium and iodine are not present in elemental form, due in part to the formation of CsI for example, and therefore do not contribute to cladding degradation from stress corrosion. There is significant stress relief in the fuel cladding as it cools down under storage conditions. Stress corrosion does not appear to be a problem in spent fuel storage.

The structural parts in fuel assemblies may consist of a number of materials including inconel, zircaloy-2 and zircaloy-4 and austenitic stainless steel. Corrosion of these parts by pure oxidation from the water at a pool temperature around 40°C can be excluded. At this temperature, electrochemical corrosion of the stainless steels can be avoided by appropriate control of pool water quality. At the end of their service life, the structural parts of the fuel assemblies are covered with an intact oxide layer which is a major factor contributing to the inhibition of electrochemical attack during storage.

Direct exposure of persons working in storage facilities can occur during such activities as handling of fuel casks and fuel assemblies, handling of contaminated filters, and repair and maintenance work. Experience shows that, in common with other fuel cycle facilities, the risk of increased occupational exposure arises when any maintenance or unusual operations are carried out. Such increased exposures can, however, generally be minimised by good planning, adequate redundancy of critical components, paying particular attention to the design of those items that are liable to become contaminated from the point of view of repair and maintenance, and by the use of local shielding and equipment decontamination procedures. Systems and components that are important in this context include:

- pool water cooling and make-up systems;
- filter equipment for purification of pool water and decontamination of fuel flasks;
- ventilation systems;
- equipment for temperature, water level and leakage measurement in the fuel pools;
- hoists and handling systems for flasks, canisters and fuel assemblies; and
- equipment for handling and storage of spent filter resins and other wastes.

There have been no serious incidents at spent fuel storage facilities involving significant radiation exposure of workers or the general public. However, some contamination of storage pools has occurred due to defective fuel or cladding corrosion, the latter particularly in the case of Magnox clad fuel storage in water whose chemistry was not controlled. Such contamination has been found to be a major source of operator dose and local spread of contamination.

Shielding is normally assured by providing a minimum of 4 metre depth of water above the fuel elements in storage, which is normally enough to reduce the dose rates to less than 0,01 mGy per hour at the pool surface. Fuel transfer mechanisms have limit switches and mechanical stops to prevent the inadvertent raising of a fuel element to within less than 3 metres of the water surface. A high-integrity pool structure is needed in order to guarantee adequate containment of the pool water, but a limited loss of water resulting in a significant reduction of the shielding layer is unlikely to involve significant risks of over-exposures to personnel since adequate countermeasures can be taken in time.

#### *Possible accidents in wet storage*

In the safety analysis of a spent fuel storage pool a number of anticipated disturbances and minor accidents have been identified. These include loss of off-site power, component failures, leakage of pool water and fuel handling mishaps. A proper design ensures that significant releases of radioactivity will not occur under these circumstances and that the resulting environmental impact will be negligible (43).

Radioactive contamination of fuel pool water can result if those fuel rods which have developed defects during reactor exposure are not identified and isolated and are therefore stored in the pools with intact fuel. Radioactive contamination of the pool water, for occupational safety reasons, has to be maintained at levels of about 30 MBq/m<sup>3</sup> or less by means of ion exchange and filtration. Additional leakage of activity from defective irradiated fuel appears to occur only slowly under storage conditions, the soluble species having already largely been dissolved in the reactor coolant in most cases.

A prerequisite for a significant release of radioactive material to the environment under more severe accident conditions is the damage of the cladding of a fairly large amount of stored fuel, with an accompanying release of gaseous and/or airborne activity from the fuel. Two conceivable mechanisms which might lead to this situation have been identified. One possibility is that the fuel overheats so that the fuel cladding loses its integrity; the other is a massive mechanical impact on the stored fuel.

A necessary condition for the former is the loss of pool water to such an extent that cooling of the stored fuel ceases and the fuel temperature increases due to decay heat. The pool water could be lost either by leakage at a rate in excess of the make-up system capability or due to failure of the pool water cooling system together with loss of make-up water capability to compensate for the evaporation which would follow pool overheating. However it has to be kept in mind that loss of water from the pool structure is inherently a slow phenomenon with only gradually increasing severity for which corrective measures can be taken in due time.

The circumstances in which an event could lead to severe mechanical loading of the fuel have been identified as:

- handling faults, e.g. fuel cask lifting device failure; and,

- external events (earthquake, tornado, flood and aircraft crash, etc.) which could cause structural failure.

Such events are taken account of in the formulation of design safety requirements. In addition to direct releases of activity following such incidents, the possibility of the subsequent initiation of other accident sequences such as criticality accidents and loss of cooling water should not be overlooked in the safety assessments.

The most severe potential accident for a wet storage facility appears to be one which involves complete loss of water from the pool due to, for example, an earthquake. To cope with this hazard a seismic design will be adopted for the pool structure, taking into account historical data and characteristics of site foundation in seismic events. Therefore an occurrence of such a severe accident is extremely improbable. Even if the cracking of the concrete pool structure was hypothesised it might not lead to catastrophic failure because any elastic return of the rebar configuration would cause the crack to close-up, to an extent, after cessation of the earthquake load. It is recognised that storage pool integrity, avoidance of large-scale mechanical damage to the fuel, and provision of adequate make-up water supply systems to cope with loss of pool cooling situations are the most important design considerations. Double-wall pool construction is now common design practice for ensuring pool integrity and it also aids monitoring for leakage. There are, however, a number of other safety-related structures and systems, which include:

- the ground housing the storage pools;
- the building structure;
- the storage pool structures;
- canisters for the spent fuel;
- emergency make-up water systems for the storage pools;
- pool water cooling systems.

If the storage period of spent fuel were to be increased significantly over presently envisaged levels, the only additional safety problems which would appear would relate to fuel assemblies during re-racking and the integrity of fuel assemblies during re-packaging and re-shipping for reprocessing or disposal.

The main considerations relating to the closer packing of spent fuel in the storage pools are: *i)*\* the storage configuration remains sub-critical, *ii)* the heat load will not raise the water to an unacceptable temperature under normal operating conditions, and *iii)* the additional weight of the fuel will not cause failure of the liner under normal and design basis accident conditions (i.e. fuel cask drop). The additional heat load will not be a significant problem if only relatively old fuel is stored in this manner.

#### *Loss of power supply*

Although the cooling plant and electrical power supply provisions normally include a measure of redundancy, this may not be essential on safety grounds alone for independent spent fuel storage installations. Immediate restoration of cooling following an interruption caused by plant failure should not be necessary and, if the plant failure condition persists, the make-up water supply should be adequate to prevent a serious fault escalation sequence from developing.

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\* For some fuel, higher density storage may result in under moderation and reduce the nuclear reactivity of the system.

Normal plant provisions include redundant components which are powered from separate bus bars so that the only common component in the AC power supply system is the main switchgear. Arrangements are also normally made for the main switchgear to be manually connected to an independent on-site power source. The on-site power supply may be a stationary or mobile diesel generator or a gas turbine installation.

A loss of off-site power for longer than a few hours has a very low probability. Loss of both off-site power and the independent on-site supply has an even lower probability and it is therefore considered that the power supply loss fault is not significant in the context of the safety of the pool cooling system. Furthermore the cooling system of a spent fuel storage pool could be stopped for a significant period (at least ten days) without any effect on fuel integrity (147).

#### *Fuel element handling faults*

A fuel handling accident such as dropping a fuel cask or canister gives rise to possible radioactive releases to the environment and may result in severe damage to the storage facility. Rupture of the cladding can occur, but if the fuel has been cooled for more than six months the only isotope of importance which may reach the environment would be krypton-85. However krypton-85 does not contribute significantly to the off-site individual exposures because it is a readily dispersed inert gas. Iodine-129 does not appear to be a problem due to the small amount of its activity in the fuel. In the unlikely event that an accident occurs with fuel which has been cooled for less than six months, iodine-131 could also be released. However iodine-131, with a half-life of eight days, would not become an environmental contamination concern. Protection of the pool structure against such faults and the provision of high-reliability handling systems are measures which should be considered.

#### *Criticality*

Criticality safety in storage pools is generally assured by the design of the baskets and racks for the fuel element bundles, primarily by diminishing neutron interaction by adequate spacing of the positions for the fuel element bundles. In many cases additionally fixed neutron absorbers, like boron steel plates, are used to reduce the water distances between the positions and to enhance the storage capacity. Consideration is also given to possible failures during loading or unloading of fuel elements, especially to drops of fuel bundles or heavy loads. Also in the event of an earthquake subcriticality of the stored fuel must be maintained. Presently the criticality safety concept for AFR pools is based on the assumption of fresh fuel, not taking into account the reduction of fissile material and the neutron absorption of fission products due to the burn-up. Recently proposals have been made for one pool to take credit of the burn-up to reach an even higher storage capacity. To implement burn-up credit at large AFR pools it would require strict control of the initial enrichment and the burn-up to avoid the erroneous loading of fresh fuel. Similarly repair or consolidation of fuel bundles within wet storage pools needs specific attention to maintain sub-criticality for all steps of operation. Compact racks and close packings of consolidated fuel may also influence the cooling requirements, the seismic design and emergency preparedness.

Special attention is given to the risk of accidental criticality which might be experienced in some fuel transport and handling operations. Some PWR fuel transport flasks contain borated water, as do the storage pools at reactor sites, and this water is usually flushed out at the reprocessing facility receipt station. Before authorising this flushing out process, or the immersion of a flask into the unloading pond at storage or reprocessing sites the water of which is not borated, a careful check of the fissile material content is made on the basis of the safety report of each container and the shipper's documents. Quality assurance procedures play an important role in this check.

Criticality safety is assured during fuel handling by applying the principle of transferring one fuel element, basket or bottle at a time. This may be achieved in ponds in a number of ways; by fixed geometry, by limitations on the permitted loads of handling systems or by redundant interlock arrangements.

Criticality accidents in pool storage systems are not likely to give rise to any release of radioactive material to the environment.

#### *Loss of coolant*

In the event of total loss of cooling, the thermal capacity of the pool water is sufficient to provide adequate cooling of the fuel for a period in excess of at least 24 hours and perhaps as long as several years, depending on the original depth of water above the stored fuel and on the quantity and the irradiation history of the fuel. It is expected that adequate time would be available to take corrective action even if complete loss of supply of cooling water occurred. As an example, a moderate water flow-rate of the order of 10 m<sup>3</sup> per hour would be sufficient to maintain a constant level in a typical pool and therefore prevent boil-off of the water and consequent cladding failure and fault escalation. However, the water flow rate for a particular spent fuel pool is dependent on the fuel burn-up, the decay time, the water volume, and pool configuration. Supplies of make-up water are usually constantly available to offset evaporation losses if it is not possible to reactivate the cooling system within a reasonable time.

Under normal circumstances the pool water temperature is kept below 40°C to maintain a habitable environment within the building without excessive ventilation. This is below the structural design wall temperature limit, which is usually set on a thermal basis. Structural damage to the main pond envelope and its liner may occur if the water temperature increases significantly beyond design limits and remains there for a long period of time. However, if such a temperature rise is only for a short time corresponding to a transient boiling accident, it will not affect the pool structure. Nevertheless, further studies to analyse spent fuel heating following loss of all the pool water have been made in order to get an insight into this extreme accident (which is beyond the design-basis accident). Among them IRS-290 (44) caused a public stir, because of a misunderstanding of the report's conclusions. However, the position of that report on such a severe accident was made clear afterwards by the same author, who made efforts to correct the misunderstanding (35).

A somewhat more comprehensive analysis has also been performed for the NRC by Sandia Laboratories to determine the temperature rise of spent fuel following a hypothetical accident involving draining of a storage pool (45, 46). Moreover deterministic studies are in progress in France on the low probability, and beyond design basis, accidents for large storage pools. In the light of these studies, it is clear that the design of a storage facility should be such that loss of coolant exceeding a few days should have an extremely low probability.

#### *5.2.1.2 Dry storage facilities*

##### *Status*

The dry storage of spent fuel is being used to a limited extent in several countries. The dry storage concepts currently being implemented include: dry wells, metal casks, silos (concrete casks), and vaults.

### *Dry wells*

In Japan, a dry well storage facility for spent uranium metal fuel from a research reactor has been in operation since 1982. The facility stores about 15 THM of such fuel. The IAEA survey (35) reported that there have been no abnormal occurrences or requirements for remedial action at the facility. In the United States, LMFBF fuel was stored in dry wells at the Idaho National Engineering Laboratory (INEL). Dry wells were used for the storage of small amounts of PWR fuel at the Nevada Test Site (NTS) as part of a large dry storage demonstration programme. Storage started at the Climax deep dry wells (600 meters below the surface in granite) in 1979. In 1983 the PWR assemblies were moved to a vault, and dry wells to the EMAD facility. One assembly underwent extensive non-destructive and destructive characterisation. No significant problems were identified.

### *Metal casks*

Designs of metal casks for use in spent fuel storage have been in existence since the late 1970s. The cask designs have capacities ranging from 4 to 26 PWR fuel assemblies and from 10 to 60 BWR assemblies. The casks are generally equipped with a double-lid system to ensure safe containment of contents. These casks have been subjected to a variety of tests and demonstrations since the early 1980s using both intact and consolidated fuel.

In Germany, full-scale demonstrations of spent fuel storage, in casks manufactured from nodular cast iron, have been in progress since 1982. In Switzerland, about 3 THM of spent fuel (in 50 per cent consolidated form) has been in storage in nodular cast iron casks since 1983. At the time of the IAEA survey (35) there had been no abnormal occurrences, however no fuel had been retrieved for inspection. Since 1984, four types of metal storage casks have been tested in the United States. The US DOE sponsored the demonstration of the storage of BWR fuel in metal casks at the Morris storage facility in 1984 and 1985. The US DOE entered into a co-operative agreement with Virginia Power, a United States' utility, to demonstrate the use of three types of metal casks. The AFR site at the Virginia Power Surry nuclear power station has been licensed by the US NRC for storage of spent fuel in metal casks. At the INEL site one case of degradation had been observed. After the first series of tests, cracks were observed in the welds of the borated stainless steel basket.

The first central AFR storage facility in Germany with a capacity of 1,500 THM uranium has been built at Gorleben. This facility makes use of the principle of dry storage of used fuel elements in transport containers of the GNS-type (CASTOR) or TN-type, in which the fuel elements are kept in a helium atmosphere to allow a long-term storage. This interim storage facility was completed in 1983 but not taken into operation. A second storage facility at Ahaus received a construction licence in 1983. Results of demonstration activities have shown the following:

- radiation and thermal levels resulting from metal cask storage have been acceptable;
- no fuel rod failure has occurred during demonstration storage;
- no secondary wastes have arisen from the storage operation.

### *Silos (concrete casks)*

Concrete cask technology has been used for the storage of spent fuel in Canada and the United States, and is also under development in other countries. In Canada, demonstration programmes for dry storage in concrete casks of commercial CANDU fuel under various environmental conditions, and of various production fuel on a routine basis, are being carried out at the White-shell Nuclear Research Establishment. Sixty-seven THM of CANDU fuel are also being stored at the Gentilly reactor site. There have been no problems or abnormal occurrences with the storage of spent fuel at these sites. In the United States a concrete cask containing PWR spent fuel

assemblies was successfully tested as part of a major dry storage demonstration conducted at the Nevada Test Site.

### *Vaults*

The United Kingdom experience with dry storage of spent Magnox fuel in vaults is extensive. The initial experience with vaults started in 1957. Three vaults were constructed for storage of Magnox fuel starting in 1962 and have been operating on a commercial basis since 1972. Two additional vaults were licensed and built and have been successfully operating since 1980. Generally, there have been no major operating problems. However, in July 1990, there was a leak of rain water into the vault at Wylfa power station; this led to superficial corrosion of 35 elements and severe corrosion of only 11 out of an inventory of some 21,000 elements (148).

France has a dry storage vault facility for interim storage of canned fast breeder reactor pins in the Marcoule reprocessing pilot plant (APM). A second facility for stainless steel clad fuel from heavy water reactors and material testing reactors has been constructed at Cadarache and named CASCAD. This facility is designed for long-term storage (50 years) of spent fuels for which reprocessing is deleted or not foreseen for technical or economic reasons. Amongst them are HWR fuels from the EL4 reactor and MTR fuels from the Osiris research reactor.

The facility consists essentially of a parallelepiped semi-buried concrete storage room, which contains 319 stainless steel tubes hanging from its roof and which are closed at the bottom. The containered fuels are unloaded from the casks, and put into baskets which are stacked in the tubes. The tubes are plugged with shielded and air-tight plugs and they are equipped with fall breakers. Air cooling is realised by normal convection. An emergency forced-air ventilation system equipped with high-efficiency filters is provided in case of air contamination or temperature inversion. This air flow enters the storage room, flows around the storage tubes and is exhausted through a stack. The design basis cooling capacity of the normal convection system is 600 W per tube while the concrete wall temperature does not exceed 80°C. The tubes are monitored for surface temperature and internal contamination. By design the building withstands the maximum safety seism of IX MSK and is sub-critical even in case of flooding. This facility started operation in June 1990. Extension of the storage facility is a reserved option.

In Japan, a dry store for about 30 THM uranium metal spent fuel was brought into operation in 1982; since then no defects or failures of storage canisters have occurred (158).

In the United States, high-temperature gas reactor (HTGR) fuel has been stored in vaults at INEL since 1964. Carolina Power and Light, a utility in the United States, received a licence in August 1986 to operate a vault type dry-storage facility consisting of horizontal concrete modules.

### *Summary of dry storage experience*

Many thousands of spent fuel assemblies of different types have been stored for periods of time ranging from a couple of years to over 30 years in more than 20 different dry storage facilities. In general the spent fuel behaviour during storage has been excellent, with the exception of the in-leakage of rain into a Magnox vault in the United Kingdom which led to only localised corrosion, and no detrimental effects of dry storage on the integrity of the spent fuel have been detected. A few leaking rods have been detected in some demonstration programmes; however the cladding defects had no adverse impact on storage or retrieval of the fuel and the rods may have been defective prior to dry storage.

### *Safety issues for dry storage facilities*

Safety issues addressed by the US NRC in their review of licence applications for metal casks and horizontal concrete modules (vaults) include: the ability of the selected materials to retain integrity in service; shielding designs and materials; fuel-basket designs and materials, including neutron absorbers; lid designs and sealing techniques; cooling duct designs (for vaults); designs for lifting hardware. Examples of site-specific issues include: storage raft/pad design and safeguards; the detection and correction of leaking cask/container lids; seismic effects on storage pads, vaults, and casks; procedures for loading and transporting casks/containers to the storage pad/vault; the radiation doses received by the workers and the public; maintenance procedures. The safety issues that are unique to concrete dry storage concepts include: the structural strength of the concrete; the ability to decontaminate the external surface of the cask or internal surface of the vault or well; the limited experience with using concrete at high temperatures.

#### *5.2.1.3 Areas of future work*

The IAEA co-ordinated research programme on the behaviour of spent fuel assemblies during extended storage (wet and dry), BEFAST Phase I, was completed and a final report was issued in 1987. The programme was evaluated by all its participants and observers as important and helpful for the nuclear community and it was decided to continue it further (1986-91) as BEFAST Phase II and BEFAST Phase III (1992-97). The areas of continued research include:

- further collection of data on dry and wet storage experience to confirm the safety of available storage concepts (including experience with consolidated fuel);
- spent fuel monitoring;
- decontamination and cleaning of spent fuel storage, transport, and handling facilities;
- storage of defective fuel (growth of defects, impacts on safety);
- long-term storage (100 years or more) in wet and dry storage facilities (extrapolation of present experience, validity of extrapolated data, accelerated tests, etc.);
- development of predictive models for the behaviour of spent fuel during and after interim storage (material aspects, failure mechanisms);
- effect of crud on spent fuel integrity.

#### **5.2.2 Fuel reprocessing**

Spent fuel reprocessing – the separation of re-usable fissile material from unwanted fission products – has been carried out on an industrial scale for more than 30 years. The separation technique which has become generally accepted is liquid-liquid extraction, with nitric acid as the aqueous phase and tri-butylphosphate (TBP) as the solvent phase, in the so-called "PUREX process".

Worldwide more than 100,000 THM of fuel in total have been processed, so far, by this technique. By the end of 1989 more than 30,000 THM of metallic fuel from power reactors (mainly Magnox), and more than 4,700 THM of oxide fuel from light water reactors, had been processed in OECD countries (47).

In France, the Marcoule plant UP1 is still reprocessing metallic fuel, whilst experience has been gained of reprocessing LWR fuels in the UP2-400 plant and, from 1989-90, in the UP3 plant (800 THM per year). A third plant UP2-800, (800 THM per year) is under construction at La Hague (Figure 5.10.1).



**Figure 5.10.1 Aerial view of La Hague reprocessing plant**

*(Source: UP3 la Hague – France)*

In the United Kingdom, the Sellafield B205 plant is expected to continue reprocessing Magnox fuel until at least 2010. The THORP (1,200 THM per year) plant is being commissioned and is due to start reprocessing oxide (LWR and AGR) fuels by the end of 1992. The Dounreay plant continues to reprocess FBR fuel.

In Germany, development of the Wackersdorf project, a 350 THM per year reprocessing plant, was stopped in 1989 (153) as well as the Karlsruhe plant at the end of 1990.

In Japan, the pilot plant at Tokai is in operation and it has already reprocessed about 600 THM of spent fuel, while a new, large (800 THM per year) industrial reprocessing plant at Rokkasho Mura is under development.

In the United States, the defence plants of Savannah River and Idaho Falls are still in operation. Reprocessing of LWR fuel, once carried out by NFS, is presently deferred.

MTR fuel reprocessing in the United Kingdom at Dounreay caters for the UKAEA's own reactors and a number of the United Kingdom and overseas customers (48). European and Japanese MTR fuel is generally reprocessed in the United States. Some was reprocessed at the EUREX plant in Italy between 1973 and 1975 (49).

All fuel cycle installations are subject to safety regulations specific to each nation, and are designed, constructed, operated and decommissioned to comply with the various regulations and

requirements. Plant design, construction, operation and decommissioning are usually controlled by licensing processes in the hands of the appropriate government regulatory bodies. Many safety-related factors have to be taken into account, in particular:

*Internal hazards*

- criticality;
- fire;
- explosion;
- corrosion;
- loss of containment due to leaks;
- loss of cooling;
- mechanical damage.

*External hazards*

- seismic events;
- extreme weather conditions;
- flooding;
- aircraft crashes;
- missiles from adjacent plant;
- fire/explosion in adjacent plant;
- subsidence potential from mining activities.

The potential risks arise from the inventory of radioactive materials which are stored and processed. One tonne of irradiated LWR fuel, one-year cooled, contains about 2 million curies ( $74 \times 10^6$  GBq) of fission products and up to 10 kg of plutonium isotopes. The hold-up in the process equipment represented the equivalent of perhaps five days of plant capacity, i.e. 5 to 7 tonnes of heavy metals (THM) per day corresponding to about 10 million curies ( $4 \times 10^5$  TBq) of fission products and about 50 kg of plutonium isotopes with a total decay heat which is much less than in a nuclear power plant.

In comparison with the quantities of radioactive material held upstream and downstream of the reprocessing plant (i.e. in the irradiated fuel store and the high level liquid waste store) the inventory in process equipment is relatively low. Nevertheless the potential risk is increased by the fact that during all stages of the process the materials are in easily dispersible forms (e.g. solutions, powders, gas) and are being subjected to sometimes vigorous chemical and physical reactions. Containment is therefore essential to restrict dispersion, in accident conditions, to the confines of the plant. The role of containment and off-gas clean up ventilation systems is discussed in detail later.

*Criticality*

As fissile material is handled in many areas of a reprocessing plant, protective measures against critical excursions are considered and instituted. Section 4.1.2 considers criticality control in general, while those aspects which are relevant to reprocessing are discussed in this subsection.

One measure is the provision of geometrically safe equipment in which, when allowing for all conceivable operational and maloperational concentrations of fissile material, reactivity will be insufficient to lead to a criticality. For cases where this would lead to unacceptably small equipment, neutron absorbing materials such as gadolinium, hafnium, and boron may be used (either fixed, or as soluble "poisons") to lower the nuclear reactivity. Other measures used for criticality control in reprocessing plants are as follows:

### *Mass control*

The total mass of fissile material in a specific vessel or section of the plant is limited, so that criticality is impossible no matter how the material is disposed. This method is applied to solids.

### *Volume control*

This applies primarily to liquids. If a vessel or piece of plant has a total volume less than the critical volume for a chemical compound containing fissile isotopes (allowing for possible blockages or overflows), then criticality cannot credibly occur. This may lead to prohibitively small-scale plant, and so one of the next two methods may be used.

### *Concentration control*

This is suited particularly to large-scale continuous processing at low fissile concentrations. For each vessel, pipe or piece of plant, a limiting critical concentration is easily derived. The criticality safety then lies in controlling the many process variables, including non-fissile feeds, to ensure that limiting concentrations are not exceeded. Generally a plant will be operated at a suitable safety margin below the various critical concentrations. In addition some in-line instrumentation is needed to monitor the fissile concentrations and profiles, and to detect unwanted accumulations of fissile materials.

### *Geometric control*

This covers a variety of techniques. It is essentially the limitation of the physical dimensions of individual pieces of process equipment so that criticality is impossible for all credible variations in the concentration, disposition or chemical form of the fissile material, anticipated for the process in question. Limited dimensions are usually based on infinite cylinder or slab geometry – where the surface-area-to-volume ratio is high – facilitating the escape of neutrons. This method is particularly suitable for processing of low-volume and high-fissile content material.

There are six main generic areas where criticality hazards are considered:

#### *The shear pack*

The irradiated fuel powder must not be allowed to accumulate to an extent sufficient to cause criticality if it accidentally becomes moderated. Before removal of the shear pack, in situ decontamination may need to be carried out. If wet cleaning is necessary, the washing liquor may require to be neutron poisoned.

#### *The dissolver*

The criticality safety of the dissolver depends on the safe geometry of the basket and/or on the addition of a soluble neutron poison to the nitric acid solution. It is more conservative not to take into account the fuel burn-up unless its control in the process feed can be ensured. When using a neutron poison, the quantities added should be controlled by two different independent methods (Figure 5.10.2 and 5.11).

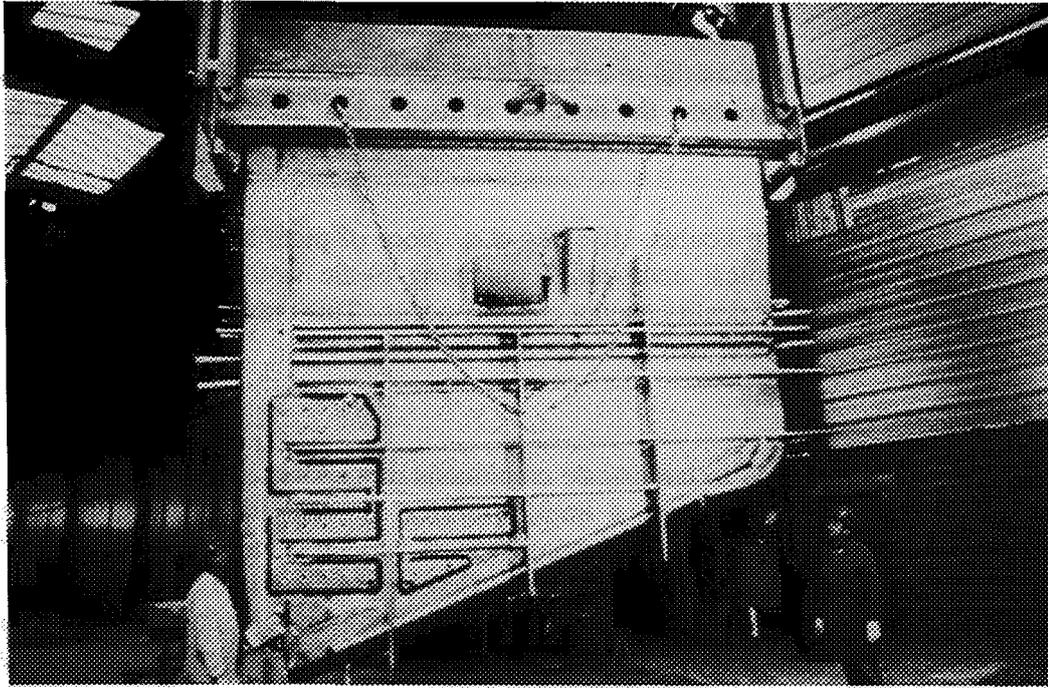


Figure 5.10.2 A view of reprocessing plant UP3 continuous dissolver

(Source: UP3 la Hague – France)

#### *Hulls storage*

The amount of uranium and plutonium left undissolved in the leached fuel hulls should be measured, mainly for accountancy purposes, before they are transferred to storage. Hull monitors select the gamma ray energies specific to certain isotopes, for example the cerium-144 and promethium isotopes. Such techniques have been shown to be satisfactory at Sellafield (the United Kingdom), NFS (Nuclear Fuel Services, West Valley, N.Y., the United States) and La Hague (France). Another technique involves neutron interrogation.

#### *The solvent extraction process*

The design aim is to make the contactors geometrically safe. If this cannot be achieved, then the uranium and plutonium concentrations are controlled, and the occurrence of conditions leading to accumulation of these elements is avoided. The required flow rate and quality of each reagent are defined and accurately measured and analysed on the plant. The uranium and plutonium concentration profiles in each contactor are carefully controlled and all equipment are regularly monitored to avoid the accumulation of fissile material.

#### *Uranium purification and oxide manufacture*

If the uranium-235 enrichment is controlled to be less than 2 per cent, criticality safety is assured without restrictions. If the uranium-235 enrichment is greater than 2 per cent, then concentration control is applied to units processing uranium nitrate solutions; geometry control is used for the oxide conversion apparatus; a combination of geometry and moderation control is used for the oxide storage facilities.

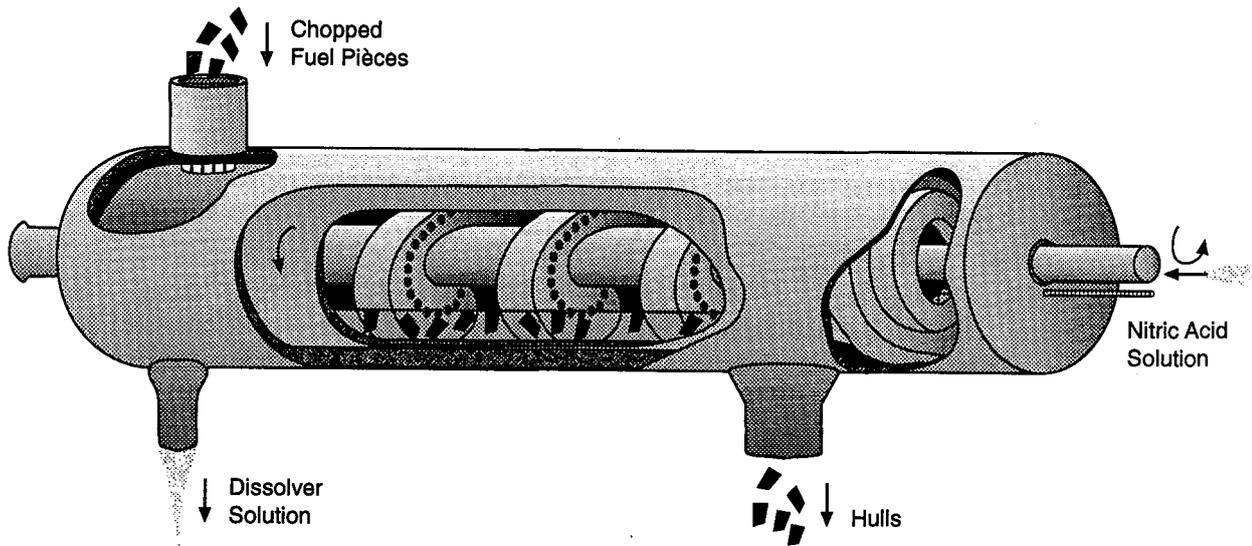


Figure 5.11 Overview of dissolver

#### *Plutonium purification and oxide manufacture*

Criticality control is essentially by geometry control, although some allowance may be made for a minimum plutonium-240 content of the material in the process.

Hazard assessments and fault studies are used to assess probable plutonium levels in these areas of plant. In particular the conditions which can lead to recycle and build-up of plutonium concentrations in contactors are determined and then avoided. Peak plutonium concentrations greater than flowsheet levels can occur, for example, as a result of temperature or acidity deviations (50). Plutonium in solution also can exhibit a tendency for forming a colloid, which plates-out and builds up on the surface of equipment if acidity decreases below a level which depends on the plutonium concentration (51, 52). These maloperations could lead to higher doses and to a mechanism by which a criticality could occur if unchecked over a period of time. Careful assessments of the conditions, good control of temperature and maintaining minimum levels of acidity should prevent the above-mentioned fault scenarios.

#### *Fires*

A sizeable fire spreading through areas of a reprocessing plant would be one of the major vectors for the dispersion of radioactive material to the environment, particularly if the ventilation system failed or the fire could not be contained within the buildings involved. A reprocessing plant handles significant quantities of combustible solvents and some potentially pyrophoric materials. Furthermore the chemical process involves the use of strong oxidizing and reducing agents. Plant safety analysis and incident analysis must allow for all potential fire (and explosion) mechanisms, in order to minimise the risk therefrom.

The risk from fire may be reduced by eliminating ignition sources and hot-spots, and by installing fire detection and extinguishing systems. The extinguishing medium should be compatible with subsequent clean-up operations as well as with the equipment and materials involved in the fire. An inerting gas may be injected into a cell but this could lead to cell overpressurisation and out-leakage; oxygen starvation by sealing the cell may therefore be a better alternative in such cases. Good reviews and prevention of major fires and explosions are contained in (53) and (54).

There are several possible combustible materials involved in the back end processes. The nuclear fuel and/or its cladding may, in certain cases, have the potential to ignite during the head end treatment. Additionally the PUREX process requires a diluent to carry the TBP through the extraction stages and this is almost invariably a flammable solvent, such as kerosene or dodecane, with flash-points of the order of 60 to 70°C. In some cases these can be ignited even when the bulk temperature is below the flash-point, if the local temperature is over the flash-point (55). Hydrogen may be used as a reductant, on an industrial scale, but it is also invariably produced as a by-product in parts of the process by radiolysis.

Special attention should also be paid to equipment in glove boxes. Plastic glove boxes present a fire risk. In the plutonium tail end process, the plutonium powder was sometimes confined in such boxes and its accidental dispersal was possible. Because handling plutonium with a high content of plutonium-238 and other isotopes makes external radiation exposure to staff a possibility, there is a trend towards remote handling of these substances via stainless-steel boxes contained in shielded cells. This results in improved containment and fire resistance.

### *Explosion*

Build-up of flammable dust, gas or vapour concentrations could pose an explosion risk. Measures to avoid accumulation of flammable gas/air mixtures, elimination of ignition sources, operation of plant below the flash point of solvents or other critical temperatures, and provision of appropriate detection and monitoring equipment can all provide protection against the risk of explosion from flammable substances. Hydrogen gas with its rapid diffusion rate, low ignition energy and wide range of explosive concentration limits, probably represents the greatest potential explosion hazard. For example, Dounreay in the United Kingdom has experienced three explosion incidents in the 26 years up to 1983, all of them associated with process hydrogen (56).

There are other possible explosion mechanisms, however. These are generally exothermic, runaway, chemical reactions in the processes which can take place under certain conditions. A thorough understanding of the mechanisms and critical parameters is a vital prerequisite to designing and operating a process within a safe regime. Two of these reactions, discussed in more detail later, are *i*) hydrazoic acid decomposition and *ii*) solvent ("red oil"), and nitric acid.

### *Loss of containment*

The number of containment barriers provided will depend partly on the assessed hazards. The first containment barrier, consisting of the process equipment and piping, may not be readily accessible for maintenance or repair after active operation has started; construction materials must therefore be selected with particular care. Much of the equipment providing primary containment in the reprocessing cycle is in contact with strong, often hot, nitric acid. In some areas there will also be highly oxidizing metallic species such as plutonium (VI) which enhance the corrosiveness of the liquors. In a few areas, particularly dissolvers and evaporators, conditions will be aggressive and erosion will add to the rate at which the materials wear away. Stainless steels, zirconium, titanium, niobium or tantalum are materials which will adequately resist those conditions. There have

been many studies which report on these material's corrosion-resistant properties; several recent articles relevant to fuel reprocessing were presented at the RECOD 87 conference (57, 58, 59).

There have recently been developments in the field of remotely controlled maintenance which have rendered feasible many repairs of active systems. Integrity of safety-related equipment is checked by periodic inspection using such techniques as television cameras, introsopes, fiber optics and pressure testing. Dissolvers and evaporators in particular merit such inspections, since they are subject to the most aggressive conditions, even though inspection may be difficult to carry out.

Primary containment may have to be deliberately breached for maintenance purposes. This breach not only results in doses to the workers but also presents a potential for release of radioactivity. A major aim in the design of reprocessing plant is the minimisation of the need for maintenance, particularly in active areas and where primary containment could be breached. There have been many developments over the past ten years, but the trend towards the use of fluidic devices with higher reliability than equipment with moving components, such as liquor pumps and ventilation valves, is of particular note (60).

Almost all this equipment is contained in thick-walled concrete cells, which form the second containment barrier. The plant building fabric may constitute a third barrier against release to the environment. These barriers are designed and constructed in such a way that the containment is not significantly degraded either by natural phenomena of even low probability (e.g. floods, earthquakes, tornadoes, etc.), or by accidental external events (e.g. aircraft crashes, explosions, missiles from adjacent plant, etc.).

The piping systems, whereby radioactive liquids are passed between different buildings of the plant, could be a weak point in the containment philosophy. Often these lines are either above ground on pipe-bridges or below ground in trenches or ducts. For this reason, special design standards have been developed in some countries; for example pipe-bridges usually require co-axial piping to provide two levels of containment, whilst trenches are liquor tight to provide a second containment barrier. Appropriate monitoring and leak detection systems can prevent accidents having a significant environmental impact.

As noted earlier, ventilation systems may often be regarded as being equivalent to one containment barrier. In a reprocessing plant there are several such systems, each designed for a specific function:

- Dissolver Off-Gas (DOG) systems are designed to handle hot and wet gases involved in, and evolved from, dissolution processes. They may be designed to remove iodine, radioactive aerosols and other volatile or gaseous species such as nitrogen oxides. Scrubbers, chemical absorbers and HEPA filters are usually favoured. The THORP plant has been designed with the ability to reduce the emission of carbon isotopes.
- Vessel ventilation systems are required to treat moist off-gases which, although drier than in the DOG situation, may also contain particulate activity. Dehumidifiers and filters are usually used in these cases.
- Cell ventilation systems are designed to handle larger volumes of relatively dry, cool air with a generally lower content of particulate activity, and filters usually suffice.
- Building ventilation systems are primarily to control environmental conditions and comfort in working areas – volume throughputs are often large, and a humidifier may be warranted.

The adequate performance of all ventilation systems is crucial in controlling the environmental consequences of any release of activity, either accidental or as an unavoidable part of the process. They must have an appropriate level of redundancy and reliability, with regular functional testing.

#### *Loss of cooling*

Generally the specific activity and heat generation per volume of active materials throughout the reprocessing stages are low and there will be a slow increase in temperatures without cooling. Only in those areas where the fission products or fissile material are concentrated will there be a substantial temperature rise in the case of loss of cooling, e.g. the centrifuge bowls of the feed clarification system. Tanks for storing concentrated highly active liquid wastes are equipped with very reliable, independent and redundant, means of cooling the contents.

#### *Mechanical damage*

Mechanical damage of components is a conceivable though remote possibility, and it could lead to release of activity under certain circumstances. Mechanical reliability of all handling systems (e.g. cranes, manipulators) is assured by good specification, design, standards and a programme of in-service testing and inspection. Nevertheless human factors are taken into consideration, particular when planning transport routes between plants and facilities.

#### *External events*

As noted earlier, external events include seismic events, extreme weather conditions, flooding, aircraft crashes, hazards from adjacent plant (missiles, fire, explosion) and subsidence potential. Plant safety against all of these is first considered at the design stage, often as part of hazard assessment studies by HAZOPS or other techniques. Essential to the quantification of the potential risk is a comprehensive database relating to the likely frequency of any particular external event. However, such databases are becoming more commonplace and of more value as utilities throughout the world have sought data in support of safety cases for new plant or studies.

Consideration of external events is essential at any early stage in the conception of a plant since they can affect the location, design and civil structure of a plant in a major way.

#### *5.2.2.1 Reprocessing of Magnox and alloy fuels*

##### *Criticality*

Since Magnox fuel comprises natural uranium, the criticality control problems are much less severe than with other enriched fuels. Consequently, there only needs to be a general mass control within the decanning caves; the dissolver can be of large capacity if required. Geometrical control is exercised over the plutonium stages, however, since the concentration of fissile material has been increased and there is a potential for accidental over-concentration.

MTR fuels, on the other hand, are basically alloys of highly enriched uranium with, usually, aluminium and are clad in aluminium. Such fuels are normally dissolved entirely in nitric acid with a mercuric catalyst, and criticality control by mass-limitation is essential. The dissolver liquors are then fed to the reprocessing plant where, as above, concentration control is exercised to prevent a criticality (5.2.2.2).

### *Fire*

The fire hazards associated with solvents and radiolytic hydrogen are generally common to PUREX reprocessing of all types of fuel. Recent articles on fire hazards and protection in general can be found (53, 54). One problem specific to Magnox fuel, however, is the potential for fires in the decanning process and subsequent fuel rod handling operations up to the stage where the fuel is put into the dissolver.

Irradiation in the reactor will develop porous outer regions on some of the rods in the core. Additionally, uranium metal in contact with water in storage ponds (if the Magnox cladding becomes damaged or perforated by corrosion) will form uranium hydride. This is an unstable compound which exhibits a strong tendency to catch fire when dry and subjected to mechanical shock. The actual decanning process, or even the subsequent jostling of a hydrated rod, can be sufficient to ignite the hydride. This fire then has a potential to propagate, particularly if associated with porous uranium, and may lead to a whole rod being consumed in a large fire since uranium metal itself will, if sufficiently hot or finely divided, burn in air.

Similarly, the presence of hydride remaining on the Magnox swarf could lead to a cladding fire. Instances of such swarf fires have been recorded in the decanning caves at Sellafield and La Hague (61). Fire detection and extinguishing systems are installed as a primary safeguard to minimise the consequences of any ignition.

### *Corrosion*

In the United Kingdom, the use of ferrous sulphamate Magnox reprocessing process leads to enhanced corrosion in some areas of the plant, besides limiting the volume reduction obtainable during evaporation of aqueous effluents.

### *General*

In 1987, the United Kingdom Nuclear Installations Inspectorate undertook a safety audit of the Sellafield reprocessing plant. The results of this audit, which concentrated on the older plant (20 to 30 years old), provide a useful view of the safety of Magnox reprocessing in particular and of fuel reprocessing generally (62).

#### *5.2.2.2 Reprocessing of uranium oxide fuels*

##### *Status*

Uranium oxide fuels are used mainly in light water reactors (LWR), pressurised water reactors (PWR), boiling water reactors (BWR), advanced gas-cooled reactors (AGR), advanced thermal reactors (ATR) and heavy water reactors (CANDU).

Reprocessing of spent  $\text{UO}_2$  fuels started in 1966 and by the end of 1990 some 4,690 THM had been reprocessed mainly in six reprocessing plants. Detailed data are presented in Table 5.2.2.2.

Table 5.2.2.2 Status of reprocessing of UO<sub>2</sub> fuel  
World record of first generation reprocessing plants

Plant	Operating period	Reprocessed fuel THM			Total
		PWR	BWR	ATR	
West Valley	1966 - 1972	131	131	-	262
La Hague (UP2-400/UP3)	May 1976 - Dec 1990	3327	762	-	4089
Tokai	Sep 1977 - Dec 1990	183.1	321.1	5.2	509.4
Karlsruhe	1971 - 1990				208
Eurochemic	1966 - 1974				180
Sellafield	up to 1973				100
<b>Total</b>					<b>5348.6</b>

Based on the experience acquired in reprocessing uranium metal fuel and uranium oxide fuel, and in order to meet the LWR fuel reprocessing demand, decisions were taken in the 1970s to build large reprocessing plants. These plants are indicated with their capacities in Chapter 9 under the heading "Reprocessing plants". According to these data, the reprocessing capacity for oxide fuel will be around 3,600 tonnes per year by the end of this century.

Spent UO<sub>2</sub> fuels have a wide range of enrichment, burn-up and cooling time. For reactor economy the tendency is to increase the initial uranium enrichment (e.g. from 3.5 to 5 per cent) to realise longer reactor cycles and consequently higher burn-up (e.g. from 33,000 MWd/t to 45,000 MWd/t). This will introduce no new problems, but will require re-examination of shielding and criticality safety provisions.

#### *Safety analysis*

There are a few marked differences between uranium metal fuel and uranium oxide fuel reprocessing from the standpoint of nuclear safety. They are due to:

- the nature of the cladding material (mostly zircaloy instead of magnesium or Magnox);
- the chemical form of the fuel (oxide instead of metal or alloy);
- the high burn-up of the oxide fuel.

Zircaloy cladding does not corrode in water pools like magnesium or Magnox so that the cooling time of the spent fuels can be extended for several years or decades, thus allowing for the radioactivity to decrease (iodine-131 and ruthenium-106 in particular). On the other hand, zircaloy fine particles arise in subsequent processing and may increase the potential fire and explosion hazards.

Due to the high burn-up of the oxide fuel a much larger inventory of fission products, though mitigated by a longer cooling time, and of plutonium and other alpha emitters has to be faced. Solvent damage and hydrazine consumption as well as the potential for criticality, radiation exposure and contamination hazards, are higher.

#### *Discussion of the head end operations*

##### *i) Mechanical operations*

The risk of fire arising from the potentially pyrophoric plutonium and uranium metals does not exist in LWR fuel reprocessing because uranium and plutonium are present as oxides. However, zircaloy chips or dust might be a potential cause of fire or explosion in LWR fuel reprocessing.

Fragmentation of LWR fuel rods by chopping with a bundle shear in the head end of a reprocessing plant produces fine zircaloy particles or even dust. Hazards could arise from zircaloy particles through self-ignition of dust layers accumulating in particular areas of the plant, or from dust explosion when particles are raised by air flows (shear housing, dissolver and centrifuge or filter).

A risk assessment programme funded by the Commission of the European Communities and the Bundesministerium für Forschung und Technologie has been carried out in Germany (63).

A systematic study of fire and explosion hazards of zircaloy fines was published by Andriessen *et al.* (64). It quantifies the parameters which influence ignition. Irradiated zircaloy fines have a bulk density of about 2.4 grams per cm<sup>3</sup>, a specific surface area of 0.1 m<sup>2</sup> per gram, a self-ignition temperature of 160 to 210°C and a glow temperature of 230 to 260°C.

Pulverised dust with a particle diameter of around 3 µm and entrained by air can explode if ignited at room temperature and in the presence of oxygen at levels well below 5 per cent. The ignition temperature, however is inversely proportional to the specific surface area of the zircaloy dust; i.e. it decreases with decreasing grain size. For dust with a particle diameter of around 18 µm, the ignition temperature rises to 360°C (64). Nitrogen gas does not react with zircaloy fines, which remain unreactive under water.

Particles size distribution shows that few fines below 100 µm are produced by shearing. Furthermore, the high proportion of inerting UO<sub>2</sub> dust, diluting the zircaloy particles by a factor of 10 to 100 within the shear, is reported to suppress dust cloud ignition (65). From a systematic study undertaken by KfK (64) and BNFL (65), it was concluded that the fire and explosion danger of zircaloy fines is not so important under normal plant conditions because of their dilution by UO<sub>2</sub> powders.

These two realistic circumstances are apparently responsible for the fact that explosions of zircaloy fines within the shear have not been reported. Nevertheless dust clouds in the shear chute are inevitable and the shear design should aim to reduce the accumulation of zircaloy fines to less than several hundred grams or, if this cannot be avoided, nitrogen inertisation should be provided to prevent potentially hazardous conditions in the shear.

##### *ii) Dissolution*

Dust clouds are unlikely, except in the shear chute where a potential for ignition cannot be excluded. These ignitions are harmless as long as the amount of fine material is restricted.

Quantitative assumptions have to be made for the safety case, and care has to be taken that inspection and removal of particle accumulations can be carried out safely.

Vessel ventilation systems are required to treat moist off-gases (cooler than in the dissolver situation), and dehumidifiers and filters are common in these systems. Cell ventilation systems are designed to handle larger volumes of basically clean, cool air. Building ventilation systems are primarily intended to control conditions in the working areas.

The performance of these systems is crucial in controlling the environmental consequences of any possible accidents; they must have adequate redundancy, be tested in situ and be regularly in use.

### *iii) Feed clarification*

Zircaloy powder and insoluble residues (metallic inclusions such as ruthenium, rhodium, palladium and insoluble fission product compounds of molybdenum, tin, antimony, zirconium, niobium) are separated from the feed solution of the liquid extraction process to prevent clogging and interfacial crusts.

For instance about 3 kg of these residues are collected from 1 tonne of LWR fuel irradiated up to 33 GWd/t. This radioactivity is still very high three years after discharge from the reactor. Self-heating of the filtered or centrifuged sludges will be insignificant in safety terms if reprocessing takes place with fuel sufficiently cooled, e.g. five to ten years.

Spontaneous heat-releasing reactions between zircaloy and fission products oxides, e.g.  $\text{MoO}_3$ , can be detected in calorimetric experiments when rapidly raised in temperature. In practice, self-heating rates are low and do not result in a violent reaction.

In conclusion, each specific safety case can only be drawn after careful evaluation of the design information and may, therefore, differ from plant to plant. Incidents can be avoided by proper assessment and design combined with operating precautions.

### *Discussion of solvent extraction operations*

#### *i) Solvent fire hazard*

All locations in the plant where sufficient solvent inventory is held to pose a significant fire hazard should be identified, and general precautions observed:

- the solvent is generally held below its flash point ( $70^\circ\text{C}$ ) throughout the process. The normal operation temperature varies between  $30$  and  $60^\circ\text{C}$ ;
- there is no identifiable source of ignition within the process plant. For instance, in-cell electric equipment is encapsulated;
- where temperatures are raised locally, for example in steam ejectors, the atmosphere consists predominantly of steam and does not favor combustion. In some situations nitrogen purging is employed.

For an in-cell fire to occur, there must be a process fault raising the solvent temperature above its flash point, a breach of primary containment, accumulation of sufficient solvent in the cell to pose a significant hazard, and an ignition source. Predicted frequencies of an in-cell fire have been assessed to be less than  $10^{-6}$  per year and no such accidents have been reported.

The possible consequences of an in-cell solvent fire have been quantified in one study (66), which can be summarised as follows:

The mechanism of combustion was that of a relatively fast primary burn in which the air available in the cell was used up. The fire would then extinguish and re-establish when sufficient air had leaked back into the cell or, alternatively, die down from the primary burn to a much slower burn rate determined by the air in leakage (secondary burn).

The extent of cell pressurisation was calculated allowing for air extraction by the engineered cell ventilation system. On the basis of these calculations the amount of air escaping from the cell during the primary burn was determined, as well as the radioactivity released. Discharge of this air and dispersion to the environment were used to determine the consequent dose uptake and risk.

The secondary burn consequences, not allowing for any fire fighting, were seen to demonstrate that the off-site risk from solvent fires was within the relevant accident risk criteria. Even so, it was concluded that good practice dictated that all cells containing bulk solvent must have some means of fire fighting. Foam extinguishing agents and/or halon inerting are preferred on the basis of seismic qualification and effluent management.

The fire-fighting equipment was, however, shown to provide benefits only in reducing the consequences of the secondary burn. Thus the risk from in-cell solvent fires is well below the criteria in all cases.

*ii) Solvent decomposition and chemical reductants*

The potential risk of explosive decomposition of solvent and degraded solvent (for example DBP) heavy metal compounds during evaporation would be increased, because of the high radiolytic degradation of the solvent in LWR fuel reprocessing, if countermeasures such as improved solvent separation from the evaporator feed solutions and limited evaporation temperature (e.g. 135°C) are not taken.

The presence of denitration reagents, such as formic acid, is a potential cause for explosion in evaporators under uncontrolled reaction conditions.

At La Hague plant, tetravalent uranium is prepared by catalytic reduction of uranyl nitrate solution with hydrogen gas under pressure. This process is also adopted by THORP. Excess hydrogen is then separated from the uranium (IV) nitrate. The main potential hazard in this operation is a hydrogen explosion. Preventive measures include a hydrogen detection and alarm system, automatic shut down of the hydrogen supply on detection of a leak, and a redundant (and possibly separate and dedicated) ventilation system designed to maintain the hydrogen concentration in the cell at below the flammable limit.

The process cells are isolated or separated from the rest of the reprocessing plant to avoid any nuclear consequence of an accident in the uranium (IV) plant. Furthermore, the equipment and cells are designed to withstand any possible explosion, to limit the consequences.

The rate of formation of radiolytic hydrogen generally increases with the radioactivity in solutions of fission products and plutonium. Therefore the risk of explosion would be higher if the vessels were not vented and, perhaps, sparged to reduce the hydrogen concentration in air below 4 per cent.

The risk of solvent fire increases also with the solvent inventory which depends upon the plant throughput.

### *iii) Explosion risks*

In 1962, plutonium reduction by uranium (IV) with an excess of hydrazine acting as a stabilizing reductant and nitrite scavenger was introduced in the PUREX process in the United States. Oxidation of hydrazine with nitrous acid or with nitrite in the presence of technetium (a fission product in spent fuel) as a catalyst leads to the formation of hydrazoic acid,  $\text{HN}_3$ , as an intermediate product.

The possible cause of explosions during reprocessing lies in the formation of hydrazine decomposition products, hydrazoic acid and azides which depend on (inter alia) technetium concentration and amounts of plutonium throughput.

Hydrazoic acid is soluble in the solvent and is back extracted by the alkaline solution in which sodium azide will accumulate. The alkaline solution is neutralised. It is acidified when it is mixed with the concentrated HLW and free  $\text{HN}_3$  might be produced giving rise to an explosion risk in the vitrification plant if it has not been destroyed previously, e.g. by nitrous acid or nitrite salt in excess.

Because of the risk of explosions of  $\text{HN}_3$ , a very conservative limit of 0.05 mole per litre  $\text{HN}_3$  was prescribed for PUREX process solutions (67). Solutions can be safely handled at much higher concentrations than 0.05M. Despite practical experience meanwhile accumulated, according to which no  $\text{HN}_3$  explosions have occurred in reprocessing plants, a critical evaluation of this safety standard in the light of recently performed systematic  $\text{HN}_3$  analyses of WAK process solutions has been undertaken in Germany (68). The results confirm that in the stream of the PUREX process a dangerous  $\text{HN}_3$  concentration does not arise.

### *Criticality*

The potential risk of criticality event in  $\text{UO}_2$  spent fuel reprocessing is increased because of residual enriched uranium, and particularly plutonium throughput and its possible accumulation by solvent radiolysis products. In order to prevent this accumulation of plutonium, appropriate measures such as routine solvent washing by caustic solution and the partial use of uranium (IV) nitrate as a reductant in the purification cycle have been adopted.

Criticality events may be caused by accumulations of fissile material not conforming to specification or by incorrect operations such as inadvertent transfer of process solutions.

Owing to the use of a favourable geometry, coupled with heterogeneous neutron absorbers and reliable fissile material monitors, no criticality accidents have been reported so far for LWR fuel reprocessing. The following protective measures may be taken against critical excursions:

- Geometrical restriction of the plant components in such a way that no criticality occurs for operational concentration of the fissile material, considering additional safety margins.
- For some components which would become too small by adopting this principle, neutron absorbing materials (for example hafnium or boron) are used to permit larger items to be used.

- Additional management controls and interlock systems are enforced where geometric control is not fully practicable. These precautions are an application of the dual failure principle under which two independent failures, each of low probability, must take place before there is a significant risk of criticality.

With respect to fuels richer in fissile material (e.g. plutonium), further R&D would assist with simplifying the safety criticality concept:

- equipment providing a higher specific throughput (fast contactors);
- reliable in-line measurement of concentration (saving vessel volume);
- computer aided process control (model improvement, model reduction);
- process parameter control to avoid plutonium accumulation in the extraction/fissile product scrub (69).

### *Corrosion*

Most of the incidents due to corrosion have occurred in the process equipment in such a way that they resulted in only minor or no releases of radioactive materials. However, plant operation is usually significantly impaired; in Tokai Mura, and at WAK, leaks in the dissolver and in the evaporator necessitated long interruptions of operation.

For instance, at Tokai, small defects in the acid recovery evaporators obliged them to suspend plant operation for about one year on each occasion for their replacement during the years 1979 and 1983 (154). Furthermore, small amounts of radioactivity were found in the steam condensate in one dissolver in April 1982 and in the second in February 1983. The corroded dissolvers were repaired in-situ by using a remotely control device. Some plutonium product evaporators have also suffered from corrosion due to REDOX processes occurring at the material surface. The welds above the liquor level tend to be those which corrode, and relatively frequent remotely control repair work has been reported in Germany, Japan and other countries. At present, the tendency is to switch to much more corrosion resistant materials, e.g. zirconium, tantalum, titanium, and 5%-tantalum alloy and chromium-25 nickel-20 steel.

Equipment which is exposed to concentrated nitric acid at high temperature (100 to 140°C) is made of 25% chromium - 20% nickel - 0.25% niobium stainless steel. Leakage failures of the acid recovery evaporators occurred after 6,000 hours and 13,000 hours of operation. They were attributed respectively to pin holes in the weld joints and to corrosion of the base metal in two heating tubes in the reboiler (70).

Preventive measures against corrosion are by careful quality assurance:

- Components are constructed and manufactured from corrosion-resistant materials under a strict quality assurance system. For example, pure zirconium for dissolvers and nitric acid evaporators; very low carbon silicon stainless steel for fission product concentration evaporators.
- The use of evaporators working under reduced pressure and at a lower boiling point constitutes a very adequate method to increase the equipment lifetime and to improve the overall safety of the distillation operation.

In addition, permanent monitoring of this first containment barrier integrity by measuring air contamination in process cells gives an early detection of leakage allowing mitigating measures to be taken.

R&D programmes are carried out on corrosion of stainless steel and examination of special metals such as titanium, hafnium, tantalum and alloys of titanium-tantalum (71), as well as further development of remotely controlled repair equipment and methods (149, 154).

#### *Exposure to radiation*

An international comparison of the exposure to external radiation of operating staff has been made (72). Despite the fact that the rates of burn-up and the throughputs are increasing, a clear tendency towards a lower radiation exposure can be recognised for the facilities still in service. In general, major contributions to radiation exposure result from repair and analytical work. It can be expected that a move towards more reliable items and the application of remotely controlled handling techniques, as well as improved automated sampling and analytical techniques, will achieve a further clear reduction in radiation exposure.

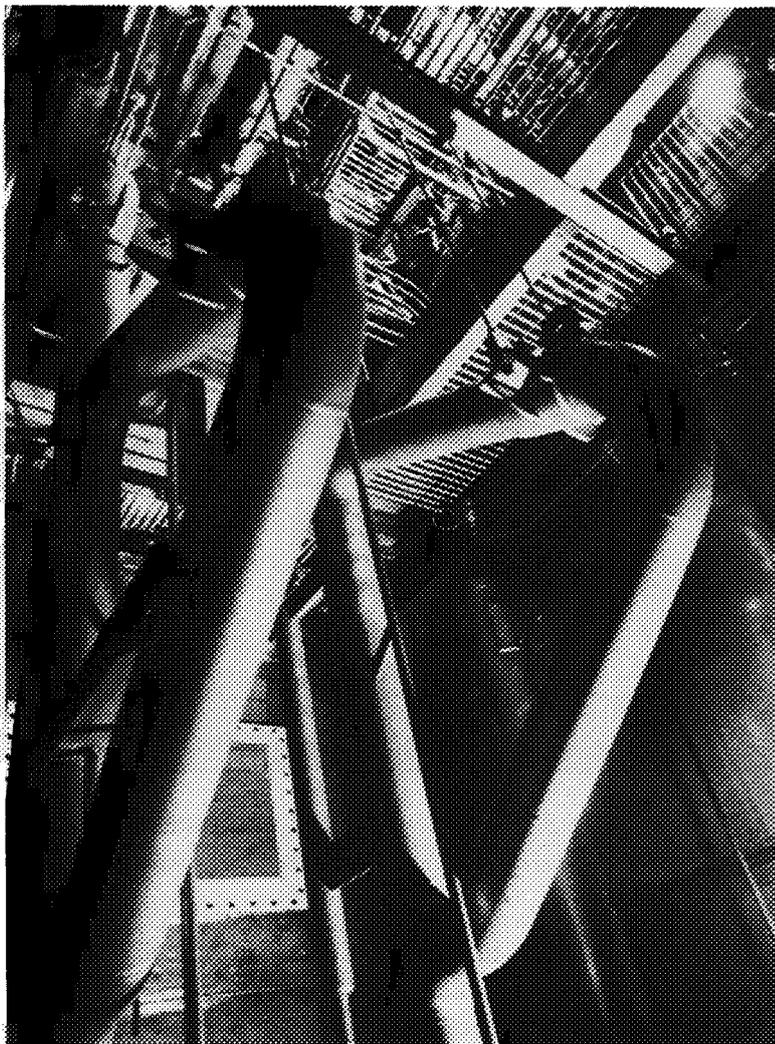
A prime objective for the La Hague plant is to reduce to nil the number of operating personnel receiving a dose higher than 5 mSv per year under normal operating conditions. This objective requires that particular attention be given to containment of radioactive materials and remotely controlled maintenance capabilities (73).

#### *Explosions in waste treatment stages*

##### *i) Waste evaporator explosion*

During operation of the solvent extraction process in the reprocessing plant, solvent may be accidentally carried over into the waste streams. The potential mechanisms for such carry-over require careful examination. The solvent and its degradation products (so-called "red-oil"), which may cause a product evaporator explosion, decompose rapidly and the United Kingdom experience has shown that tributyl phosphate (TBP) in the waste stream is converted to phosphoric acid within a matter of hours. Red oil explosions can be prevented by eliminating the accumulation of organic materials in the waste, and by controlling the process temperature in the evaporator.

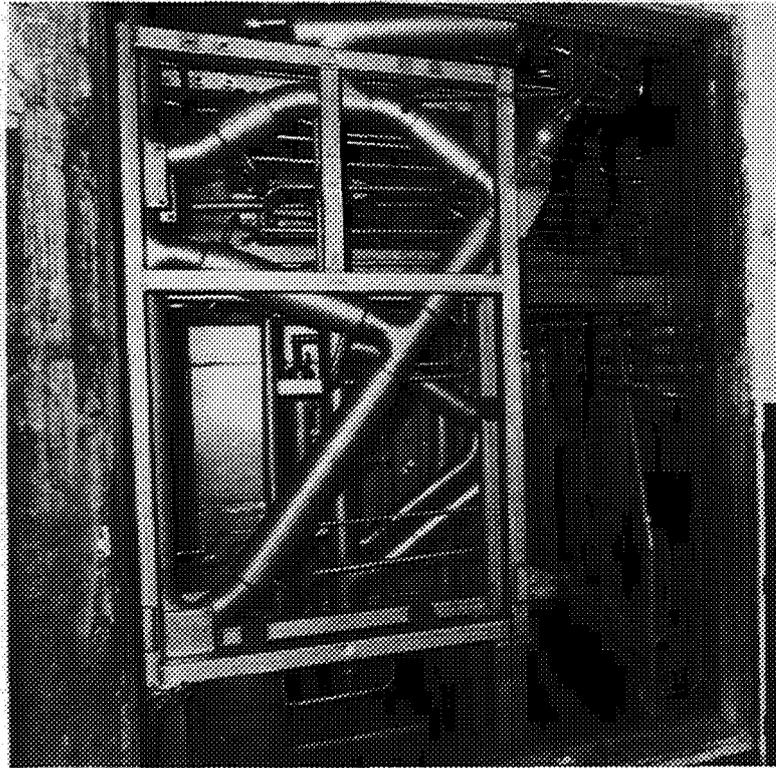
The solvent entrained and dissolved in the waste stream is either washed away by a counter-current stream of diluent, e.g. at La Hague, or is carried away by a counter-current flow of steam, e.g. at Sellafield. The evaporator is operated under reduced pressure in order to lower the boiling point of the liquor and decrease corrosion rates and to ensure that the temperature is well below the threshold for solvent/nitric acid reactions. The TBP solvent now used is considerably less reactive with nitric acid than the Hexone used in the first American plants and the Butex used in the first United Kingdom plant.



**Figure 5.12 Overview of evaporators**

*(Source: Downreay AEA Technology – The United Kingdom)*

When formaldehyde is added to destroy nitric acid (e.g. in the French process) the reaction conditions (temperature and acidity of the solution, pressure in the evaporator, feeding rate, etc.) are adjusted so that the reaction will be kept under control and proceed smoothly. The initiation phase of the reaction is the one that requires the greatest attention, and the addition of formaldehyde is adjusted to prevent the accumulation of unreacted reagent which could lead to an explosion.



**Figure 5.13 View of in cell equipment**

*(Source: Dounreay AEA Technology – The United Kingdom)*

Waste evaporators are installed in highly shielded cells (typically 1.5 to 2.0 m thick concrete). The lower part of the cell is lined with stainless steel to contain any spillage. The cell is equipped with ejectors to transfer the liquor to alternative vessels. One or more additional cells, which may or may not initially contain a second evaporation unit, are usually provided to act as a spare in the event of a repair being carried out on the original evaporator. A rolling programme of waste evaporation refurbishments may be required to maintain a continuous evaporation capacity. The highly active evaporator is a thick-walled stainless steel vessel, usually a pot type built to a well-tested design which has been proven on the existing plants.

A waste evaporator explosion could conceivably breach the primary containment barrier depending on accident scenarios, but would almost certainly not affect the secondary containment system. Although waste evaporator explosions have not been a problem, in the United Kingdom it has been calculated that, for the Sellafield plant, an explosion could release about 600 litres waste solution into the cell as a finely divided mist. A substantial fraction of the mist would rain-out or plate-out on the cell surface. The droplets remaining in the air would be carried through the ventilation ducts to the high-efficiency particulate air filters (HEPAs). Moisture separators upstream of the filters would remove most of the mist to substantially limit the activity released to the environment (74).

*ii) Plutonium evaporator explosion*

An explosion in the plutonium evaporator in the reprocessing plant might develop into one of the worst possible potential accidents by which plutonium could be released. Preventive measures against such events are described in 5.2.2.5.

### *Ozone explosion in cryogenic krypton removal*

Since radiolysis of oxygen leads to the formation of ozone, control procedures may be required to keep its concentration low. If a krypton removal stage is ever required, the cryogenic distillation process to be used for this purpose would bring a potential ozone explosion hazard.

Assessment of the ozone explosion hazard in the krypton removal from the dissolver off-gas by cryogenic distillation has been undertaken (75). The results show that the explosive limit is nearly independent of the type of noble gas and approximately proportional to the heat capacity of the gaseous diluent. Also a reduction in rectification column diameter causes a slight increase in the explosion limit and a substantial inhibition of the transition from a deflagration into detonation. A pressure increase of the explosion mixture leads to a reduction in explosive limit.

An alternative process has been developed in Karlsruhe, which involves nearly complete removal of oxygen from the feed-gas by catalytic reduction (76). Krypton removal techniques based on absorption in fluorocarbon solutions at cryogenic temperatures have also been developed (77).

### *Operational experience*

#### *i) UP2-400 plant*

The good operating record of the La Hague UP2-400 reprocessing plant is mainly due to:

- the great attention paid to career-long training of personnel;
- the plant operators' responsibility for safety under the safety authorities' control;
- the constantly improved technology, particularly for mechanical operations in the head and tail end shops which were, in the early years, the cause of most plant shutdowns;
- the experience based on preventive maintenance as well as the rigorously prepared and performed maintenance work and strategic interventions. For example, there was an intervention from November 1985 to January 1986 on the centrifuge contactor in the HAO head end; the beta-gamma dose rate measured on the shaft was 140 Gy per hour, the actual collective dose for the 70 people concerned was only 23 man mSv (78).

The plant is operated at its nominal capacity of 400 tonnes uranium per year. End products (uranyl nitrate and plutonium dioxide) were recovered with high yields while meeting the requisite specifications.

Reprocessing plant performance in terms of throughput and availability can only be obtained by achieving high safety standards. For UP2-400, no significant incident in operation has been reported. The annual average individual dose rate of La Hague personnel was 2,13 mSv in 1985.

Radioactive discharges have been in compliance with Regulatory release limits under all circumstances. A reduction by a factor of 2 of the activity discharged to sea was obtained in 1989.

### ii) UP3 plant

For UP3, major changes were made to improve the PUREX process performance, such as:

- measures to prevent dispersal of alpha emitters to most liquid streams and to confine them to those effluents to be vitrified;
- improvements concerning decontamination performance (neptunium, technetium, etc.) and the reduction of the salt content in liquid and solid wastes;
- complete recycling of recovered reagents – nitric acid, water, TBP and diluent.

Process equipment was designed to achieve increased safety and reliability by:

- use of new materials, e.g. zirconium to avoid corrosion problems with hot nitric acid (in dissolvers, and also in evaporators for acid recovery);
- geometrically safe annular and cylindrical pulse columns with improved packing (79);
- whenever possible the choice of continuous processes over batch processes, e.g. a larger capacity, geometrically safe, continuous dissolver and high-capacity, continuous plutonium-oxalic conversion equipment).

### iii) THORP

Improvements were also made for THORP such as:

- adoption of salt-free reagent to effect plutonium/uranium separation (compared with ferrous sulphamate reductant) to produce salt-free, medium-active effluents for evaporation;
- whole oxide fuel shearing compared to decanning of metal fuel;
- incorporation of a centrifuge to remove insoluble fission product alloys from HA dissolver liquor;
- uranium/plutonium co-decontamination, separation and purification undertaken in pulsed sieve plate columns rather than in mixer settlers (Magnox). This change enables the plant design to avoid criticality and to minimise liquor volumes, residence times and radiation damage to organic solvent, thus minimising potential plutonium losses in some effluents;
- conversion of plutonium nitrate to  $\text{PuO}_2$  product will utilise the oxalate route of precipitation, filtration and calcination in continuous (rather than batch) operated equipment (80).

### iv) TOKAI

The Tokai plant is a typical pilot reprocessing plant which has a good operational performance. The main characteristic is the low level of radioactive sea discharge.

#### 5.2.2.3 Mixed oxide fuels

##### *Status*

Only small quantities of MOX fuel (about 3 THM) have been reprocessed in pilot plants at Karlsruhe (WAK) and at Marcoule (APM) – (81, 82). A first experimental MOX fuel reprocessing campaign is planned in the UP2-400 plant involving 8 THM.

Table 5.2.2.3 shows the expected medium-term evolution in spent fuel from LWR in France. MOX fuel is taken as the reference fuel for the UP2-800 plant which has two shearing and

dissolution lines. This plant will be operated when MOX fuel is reprocessed with the two lines in parallel, one processing MOX fuel and the second standard LWR fuel, the dissolution liquors being mixed before the solvent extraction stage (83).

### *Safety analysis*

Compared with standard LWR fuels, high burn-up UO<sub>2</sub> fuel and MOX fuel present significant changes in the safety assessment:

- Increased plutonium flows and criticality risk.
- Higher neutron emission from spent fuel. Second generation MOX fuel will emit 35 times as many neutrons as standard LWR fuel, therefore it is diluted with standard LWR spent fuel although additional neutron shielding is still required.
- Higher alpha power by a factor of 15 between second generation MOX fuel and standard LWR fuel; this will reduce the effectiveness of the cooldown storage period before reprocessing.
- Higher plutonium-238 content: 1.7 per cent for standard LWR fuel; 2.8 per cent for high burn-up UO<sub>2</sub> fuel; 3 per cent for first generation MOX fuel; 3.3 per cent for second generation MOX fuel made with depleted uranium; and 3.9 per cent for second generation MOX fuel made with reprocessed uranium. The outcome will be significantly increased radiolysis in all stages of solvent extraction but especially in the final plutonium purification cycle and in the plutonium oxalate precipitation.
- More difficult dissolution of MOX spent fuel might necessitate more aggressive conditions and result in further corrosion and criticality considerations; this can be alleviated by using carefully controlled methods for initial production of the MOX fuel.
- Waste production has to cope with higher alpha activity content.

Table 5.2.2.3 Expected medium-term evolution in LWR spent fuel from EDF reactor in France

	High Burn-up UO <sub>2</sub> Fuel	First Generation MOX Fuel	Second Generation MOX Fuel
Enrichment (uranium-235 or total plutonium)	3.7%	5.25%	7.4%
Average burn up	42,000 MWd/T	33,000 MWd/T	42,000 MWd/T
Maximum fuel assembly burn up	49,000 MWd/T	39,000 MWd/T	49,000 MWd/T
Date of reactor loading	1988	1987-92	1992
Refuelling (1/3 or 1/4 core)	1/4	1/3	1/4
Cooldown before reprocessing	5 years	3 years	5 years
Start of reprocessing	1997	1993	2001

#### 5.2.2.4 *Fast breeder reactor (FBR) fuel*

##### *Status*

Some 100 TMH of FBR fuel have been reprocessed in pilot plants after various cooling times (with a minimum of 136 days). In addition, a campaign of FBR fuel has been carried out in UP2-400. The dissolution liquor was mixed with a GCR fuel dissolution liquor before feeding to the extraction process. Detailed data on FBR fuel reprocessing are shown in Chapter 9.

##### *Safety analysis*

The principal differences between FBR and thermal reactor fuel reprocessing concern the physical structure of the FBR fuel element and the problem of disassembly, for access to fuel pins, in the face of high-heat ratings. The solution adopted is to open up the stainless-steel fuel element wrapper (e.g. by laser cutting) and to withdraw individual fuel pins for single or bundle chopping and loading to the dissolver basket. Thereafter, the separation chemistry operations are constrained by the high plutonium content of fuels. This demands the use of geometrically limited plant to avoid criticality which, in turn, places limits on the maximum HM throughput.

##### *Experience in operation*

Reprocessing of FBR fuel, particularly after a short cooling time, requires very specially designed and engineered facilities in order to cope with the very high plutonium content, the high-heat output fuel and the potential presence of sodium on and in the fuel pins. Up to now, the chop operation has been carried out on single pins (or small bundles) in order to avoid excessive heating of the equipment. The presence of residual sodium may require inerting of the internal chopper space to avoid sparking and incidental hydrogen explosions after contact with aqueous solutions or moist atmospheres.

In the UP2-800 plant the possibility of handling plutonium fuel with an initial enrichment as high as 10 per cent has been studied. The main conclusions are that such reprocessing is feasible after cooling of the FBR fuel for at least five years and then using dilution with standard LWR fuel. The FBR fuel subassemblies would have to be disassembled in a special workshop, as was the case for the FBR fuel reprocessed at UP2-400, so as to feed the shear with bundles of fuel pins in containers.

The Dounreay pilot reprocessing plant for FBR fuel continues to treat FBR fuel from the United Kingdom reactors but the prospects of enlarging the plant are rather slim.

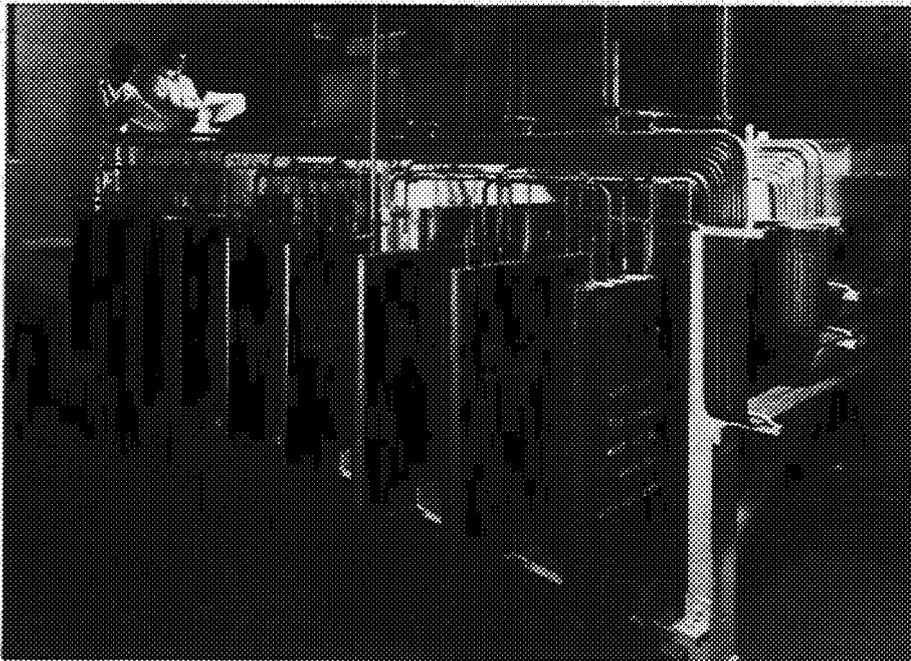
The Tokai test plant is reprocessing the fuel from the Japanese experimental fast breeder reactor. Japan has a national programme for the future development of the next generation of equipment for fast fuel reprocessing plants.

#### 5.2.2.5 *Plutonium storage and conversion to oxide*

##### *Liquid evaporation and storage*

One end product of the separation process is a solution of plutonium nitrate at a concentration of a few grams per litre. This solution is unsuitable both for economic storage and for any subsequent stages of oxide or metal production. In the United Kingdom it is therefore concentrated by evaporation to about 300 grams plutonium per litre. The plutonium nitrate is stored in buffer

tanks (Figure 5.14) prior to its transfer to an oxide conversion plant. In France there is no storage of this plutonium nitrate solution but direct in-line conversion to PuO<sub>2</sub> for storage or transportation.



**Figure 5.14 Plutonium liquid storage**  
(Source: Downreay, AEA Technology – The United Kingdom)

An explosion in a plutonium evaporator in the reprocessing plant would be one of the worst possible accident scenarios (depending on the plant design) in which plutonium could be released to a cell, a glove box area, or even to the environment in a severe case. Solvent may be accidentally carried over in aqueous waste and product streams. If solvent and its degradation products are heated in an evaporator, in contact with nitric acid and heavy metal species, they decompose and under certain conditions the decomposition may conceivably enter a thermal runaway stage and over-pressurisation, or even an explosion, could occur. Although only a remote risk, these runaways and explosions can be prevented by *i*) minimising the accumulation of organic materials in the plutonium stream fed to the evaporator by washing and *ii*) controlling the process parameters in the evaporator. Temperature has the principal effect and is generally maintained below 135°C (61), for safety and corrosion-rate reasons.

When hydrazine is used in the process there is a remote possibility of hydrazoic acid arising in the evaporator, with a potential for an explosion. A hydrazine destruction stage, usually using nitrogen dioxide gas injection, may be used prior to evaporation of the product. This potential explosion scenario has been studied recently and deemed to be inconceivable (47).

Evaporator explosions could breach the primary containment barrier but would be unlikely to affect the secondary containment system. Any aerosols generated in cells would either plate-out or be carried through the ventilation ducts to be removed by the HEPA filters. There is a risk that the explosion shock waves could over-pressurise and damage the filters. However, if the filters are distant from the cell then the energy is likely to have been mostly attenuated by the ductwork before reaching the filters. Cells are lined in stainless steel to contain any spillage or leak. Leakage from

tanks and evaporators are avoided by good design, use of resistant materials and strict Quality Assurance.

The generation of hydrogen, and sometimes oxygen, by radiolysis can be relatively rapid in concentrated solutions of plutonium nitrate. Explosive hydrogen/air mixtures could build up and it is normal practice to maintain a sparge of air through storage tanks to keep the hydrogen content below the lower flammable limit of about 4 per cent.

The potential for a criticality excursion is present throughout the plutonium nitrate evaporation and storage plant. Prevention is generally by geometrical control (see 5.2.2.2); tanks for plutonium nitrate solutions are generally either thin slab tanks, or "harp" tanks composed of rows of geometrically safe pipe.

In the presence of strong nitric acid, some plutonium (VI) may arise and this is a particularly strong oxidizing agent which may enhance the corrosion of the primary containment. Corrosion conditions within plutonium evaporators are therefore aggressive and there has been much research devoted to suitable materials in order to minimise the risk of leaks (58, 59, 163). Zirconium, titanium and a number of stainless steels are all potential candidates. In France, zirconium has generally been the choice, based not only on its resistance to general corrosion but also on its insensitivity to intergranular corrosion (25, 164). In the United Kingdom, titanium has generally been used for plutonium evaporators in the past, but THORP will use zirconium.

Plutonium nitrate solution is transported in a flask carried by boat from Dounreay to Sellafield in the United Kingdom, for conversion to oxide. Clearly, the above hazards have to be carefully considered for such movements of fissile material.

#### *Plutonium conversion and storage in solid form*

##### *i) Conversion process*

The purified plutonium nitrate solution is received in a buffer tank before feeding to the conversion line in which it is transformed into  $\text{PuO}_2$ .

In general, plutonium is precipitated from the nitrate solution by oxalic acid at a temperature of about  $60^\circ\text{C}$ . The plutonium oxalate precipitate is filtered and washed; the supernatant liquor is recycled in the process. The precipitate is transferred to a calciner furnace where it is first dried at a temperature of about  $250^\circ\text{C}$ , then converted to  $\text{PuO}_2$  by calcination at a mean temperature of  $450^\circ\text{C}$ .

The  $\text{PuO}_2$  powder is transferred from the furnace to a hopper/mixer. Homogeneous  $\text{PuO}_2$  powder is then loaded into stainless-steel cans directly connected to the hopper/mixer. The cans are disconnected from the hopper/mixer, closed with a lid and loaded in a stainless-steel transfer jacket. The jacket is closed with a welded cover. All these operations are carried out in glove boxes. The jacket is disconnected from the glove box without any breach of containment and put into a stainless-steel container for shipment to storage. The plutonium oxide containers are stored in a vault building.

##### *ii) Safety problems*

Tanks containing plutonium solutions are vented and may be sparged with air in order to maintain low, safe, radiolytic hydrogen levels.

Containment of the very fine  $\text{PuO}_2$  powder is a major concern in operation. In this respect the development of a stainless-steel jacket concept replacing the sleeves made of plastic material for the transfer of  $\text{PuO}_2$  cans is a decisive improvement. The workers are generally protected by at least two containment barriers: process equipment and glove box in the conversion plant; storage channel and container in the storage building.

Criticality is another major concern, therefore all process equipment in the conversion plant and store are assessed and designed to be geometrically safe.

For the store it is necessary to demonstrate that storage channels and  $\text{PuO}_2$  containers are designed to withstand the maximum rise of pressure from radiolysis and heating (several bar) without loss of containment.

All appropriate process equipment of the conversion plant and channels/containers of the storage building must also be designed to withstand the maximum rise of temperature which may occur by plutonium heat decay in case of extended loss of cooling.

The conversion plant and storage building are designed and constructed to remain subcritical in case of flood, fire and earthquake.

#### *5.2.2.6 Uranium storage and conversion*

The major end product of the separation process is a solution of uranyl nitrate at concentrations of a few tens of grams per litre. The volumes of uranyl nitrate produced by commercial reprocessing are such that storage in liquid form is far from an economic proposition. The liquid is invariably concentrated by evaporation, to typically 400 to 500 g/litre, and maintained in buffer prior to conversion to uranium oxide power in a thermal decomposition process.

If the uranium-235 enrichment is controlled (to be less than about 2 per cent), by blending with depleted uranium if necessary, then criticality safety is ensured without other restrictions. Calculations will confirm the precise limiting figure for a particular set of plant and process parameters. If there is more than about 2 per cent uranium-235, concentration control is normally applied (see 5.2.2.2) to uranyl nitrate solutions; geometrical control is used in conversion to the oxide and for oxide storage facilities (occasionally in conjunction with moderation control).

The handling of recycled uranium poses additional safety problems in particular that of enhanced radiation dose rates. This has been covered in some detail in 5.1.3.6, *ii*).

The red oil hazard, referred to earlier in 5.2.2.5, also needs to be considered for the uranium liquor evaporator. The radiolysis problem is much less severe. Conditions in uranium evaporators are generally less aggressive than in plutonium evaporators, and stainless steel may be used for their construction.

Uranium oxide may be prepared by direct thermal denitration (TDN) of uranyl nitrate, or by precipitation as ammonium diuranate (ADU) followed by calcination. In the TDN process, uranyl nitrate solution is sprayed directly onto a fluidised bed of  $\text{UO}_3$  powder. The material of construction for the vessel requires careful selection to avoid erosion problems.

Both TDN and ADU processes generate finely divided uranium oxide powder, the handling of which requires special ventilation provisions to prevent contamination of, and ingestion by, personnel.

Uranium oxide is stored in mild steel drums and generally presents no hazard. Radiation levels are very low for low burn-up material, but the presence of uranium-232 (section 5.1.3.6) in product from high burn-up fuel results in significant gamma radiation levels, which increase with time to reach equilibrium after about ten years. Storage facilities for such material have to take account of these features and hence are designed to higher standards than some existing facilities.

### **5.2.3 Radioactive waste management**

#### *General background*

The management of radioactive waste requires the adoption of a number of practices and policies, all with the same objective, i.e. the handling, interim storage and disposal of radioactive waste with all reasonable safety precautions to prevent subsequent undue radiation exposure of man or contamination of the environment in both the short and long term.

Many reports have recently dealt with the question of disposal, notably at the international level (84, 85), and consequently the safety of waste disposal is not included in the scope of the present report. However, the accident potential of the operations of handling, treatment and storage of radioactive waste prior to disposal has to be considered, as with any other operation involving radioactivity within the fuel cycle. Waste practices are therefore reviewed in this section in an order which to some extent reflects their relative significance since it broadly corresponds to a decreasing level of radioactive contamination.

Some minor process wastes arising from wet storage and spent fuel dissolution, such as ion exchange resins from pond water treatment, barium carbonate (containing  $C_{14}$ ) and silver zeolites (containing  $H_3$ ), have to be mentioned for completeness. Their conditioning (e.g. in polymer or concrete) and storage do not give rise to any significant safety issues and they are not discussed any further.

Overall the processes, however, are designed to minimise the quantity or volume of waste generated and their radioactivity, and to provide safe and easily retrievable on-site storage of those wastes held on site. Considerable R&D efforts have been devoted to exploring new techniques for waste minimisation (see, for example 151, 152, 155, 161, 167).

#### *5.2.3.1 High level liquid waste*

High level liquid waste (HLLW) arises from the reprocessing of spent nuclear fuels, which are dissolved in acid, and the plutonium and unburned uranium removed in the chemical separation plant. The HLLW is a mixture of nitrate salts in nitric acid containing more than 99 per cent of the non-gaseous fission products, very low levels of uranium, plutonium, but virtually all the other transuranium elements originally present in the spent fuel. It also contains activation products, corrosion products of iron, nickel and chromium, and may contain other chemicals which may have been added during reprocessing such as sodium, gadolinium, cadmium, boron and fluorine. All HLLW requires cooling.

After its separation during reprocessing, high level liquid waste is concentrated by evaporation to minimise the number of storage tanks required. Liquid storage is now usually regarded as an interim step between reprocessing and solidification, but the time scale and duration of liquid storage is dependent upon many factors. The cooling period of spent fuel – at least 90 days – is normally governed by the decay of several nuclides, e.g. ruthenium-106 and iodine-131. In practice, reprocessing of LWR fuel takes place only after a few years of cooling (the reference storage time of spent fuel before reprocessing is presently 3 to 5 years). This delay allows many of the short-lived fission products to decay and the specific activity to lower, reducing the problems of reprocessing. Similarly, the eventual solidification of the liquid waste may be easier after prolonged cooling.

In theory, other factors may also have to be taken into account, such as the commercial interest in the extraction of certain fission products from the liquid waste solutions and the possible separation of actinides. Such activities presently do not seem likely to be pursued on the industrial level in the near future. The current practice is to store high level liquid waste in stainless-steel tanks with forced cooling. Failure of the storage tanks and particularly of the cooling system could potentially result in the release of fission products in large quantities, and this form of storage requires continuous surveillance. This is why solidification is regarded as a significant improvement in terms of safety although the tank storage system has a satisfactorily high level of safety, as has been shown by about forty years of safe operation.

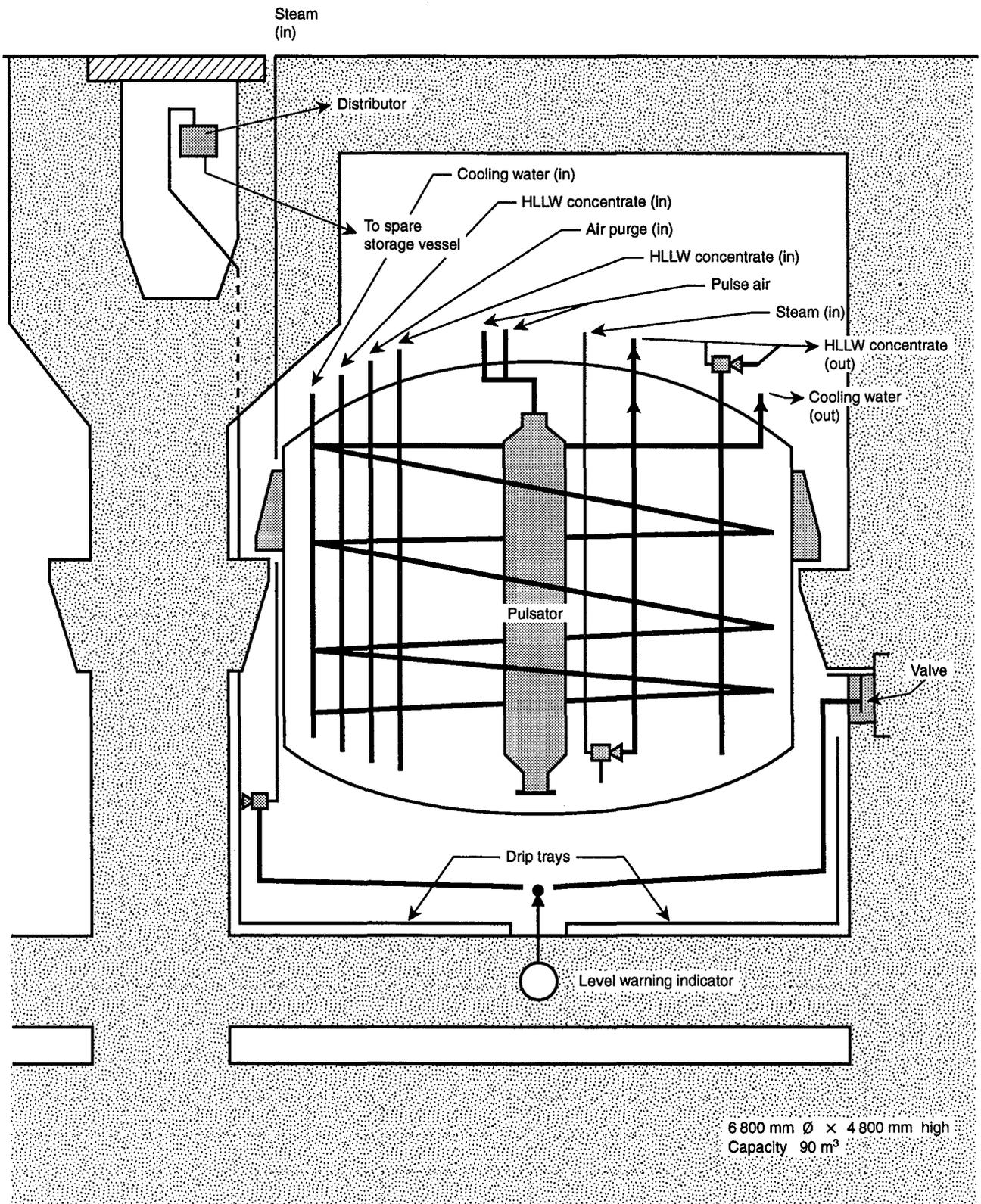
### *Liquid storage*

#### *i) Technology and experience*

Spent fuel reprocessing and high level liquid waste storage began about 40 years ago and therefore, relatively long-term experience is available. This experience was reviewed by the IAEA in a technical report and it still constitutes the state-of-the-art (86). HLLW is now usually stored as nitric acid solutions in high-integrity structures; these are single or double-walled tanks, constructed in stainless steel, and located in thick-walled concrete cells which provide radiation shielding and additional containment barriers. A typical HLLW tank contains about  $10^7$  TBq in  $100\text{ m}^3$ . To remove the decay heat generated, storage tanks require a complex and reliable cooling system. The tanks must also be equipped with some form of agitation system to prevent any precipitates from settling and creating local corrosion spots, and there is provision for transferring the contents to a standby tank in the remote event of a leak. In this respect, a spare tank policy is usually established to cope with the possible failure at any time of any of the tanks in use. Systems are also provided for venting and off-gas treatment and monitoring of temperatures and liquor levels. The design of an HLLW storage tank used in Japan is shown in Figure 5.15.1 and a photograph of a HLLW storage tank under construction is shown in Figure 5.15.2.

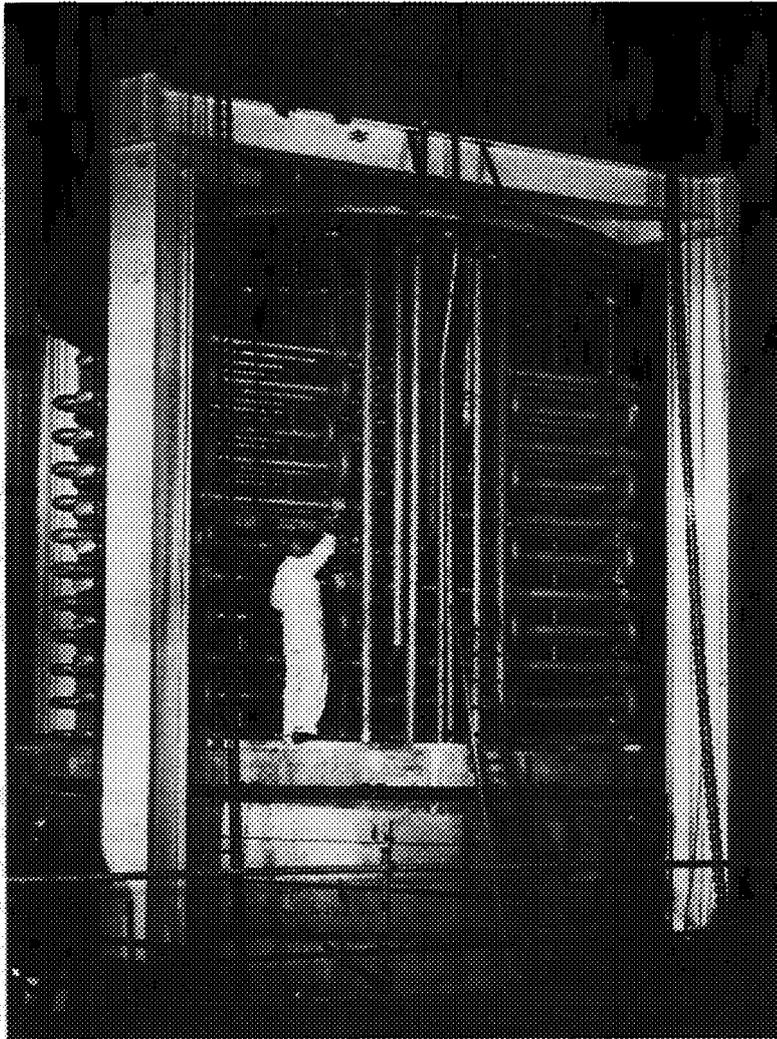
At present, the tank sizes in use, or projected, range from  $50$  to  $150\text{ m}^3$  capacity. The tanks are located, either singly or in groups, in concrete cells  $1.5$  to  $2.4\text{ m}$  thick (it is possible for them to be built partially, if not wholly, below ground level). This thickness of concrete provides shielding and protection against external mishaps such as an aircraft crash. The floor and walls of the cell are lined with stainless-steel to a height sufficient to contain the entire contents of at least one tank in the unlikely event of a leak. A gradual slope in the cell floor leads to a sump, and cell wash-down pipework is included for the event of a leak from the stainless-steel tank. The design of the tank cooling system, support structure and the enclosing cell should take into account the possible rise of the ground water level as well as other extreme natural events such as earthquakes. Special drainage and construction may therefore be required, and details vary with site and facility design.

Figure 5.15.1 Sketch of High-Level Liquid Waste (HLLW) Storage Tank (PNC, Tokai)



The cooling system consists of a series of separate internal coils arranged in a number of parallel circuits. They have sufficient heat removal capacity to cope with the maximum thermal output envisaged for the tank liquor, plus some spare capacity in case of failure of one or more circuits.

In addition to coils inside the tank, some tanks are equipped with an outer cooling jacket which can have two other uses: acting as a further line of containment if the main tank leaks and helping to keep the bottom of the tank cool and prevent increased corrosion if solids settle on the tank bottom.



**Figure 5.15.2 View of HLLW tank with cooling coils**

*(Source: BNFL – the United Kingdom)*

In all cases, it is important that reliable sources of cooling water are available. Cooling ponds, cooling towers, wells, rivers and oceans are all potential sources. In the case of a large lake or ocean, one source with multiple transfer lines may be sufficient. At the NFS plant in the United States, and in France, air-cooled finned heat exchangers provide back-up cooling.

Most installations used closed-loop cooling water systems between the tanks and the external cooling water source, separated by heat exchangers. Thus radioactive materials which might be released from a leaking cooling coil would not be transferred to the external cooling water. The provision of a small, positive pressure differential between the cooling water and the tank's contents would also reduce the chance of activity entering the cooling water in the unlikely event of a leak in a coil.

The cooling water supply system incorporates a standby source of electrical power for the circulating pumps. The reliability of all power supplies is assessed and proved satisfactory.

When a new tank is being brought into use after commissioning, it is usually partially filled with water or dilute nitric acid to ensure that enough of the heat transfer surface is covered with liquid to remove the decay heat as the waste solution is fed into the tank. Once the normal operating level has been reached, the excess liquid is removed by in-tank evaporation and the volume available is filled with fresh waste solution. Careful control of the rate of addition of fresh waste is essential, together with monitoring of the temperature and heat removal. The operation of the cooling system, involving each cooling coil circuit as well as the outer jacket (if provided), is monitored to ensure that design heat transfer coefficients are maintained; one of the early effects of sludge formation in the tank is a local decrease in the heat transfer coefficient and consequent local temperature rise.

To minimise corrosion, temperatures should be kept below 60 to 65°C. Any differences in temperature profile, particularly from top to bottom, will indicate sludge formation at the bottom of the tank.

A great deal of experience has been accumulated in several plants throughout the world on the operation of high level liquid waste tanks. Leaks from the HLLW solution to the cooling circuit have been observed and have led to the shut off of the cooling coils from the system. In normal operation, safety is based on the usual radiation protection and containment barrier principles associated with good engineering practice, highest quality of construction and continuous monitoring and control of the important parameters: temperatures and liquor levels.

#### *ii) Safety problems*

To provide a safe high level liquid store requires reliable containment and cooling, and adequate radiation shielding. A project safety assessment must include studies of site ecology, geology, seismology, hydrology and meteorology, and an evaluation of the impact of the site on the local environment. The studies must also include information on the frequency and severity of local natural hazards such as earthquakes, flood, tornadoes and hurricanes, so that due allowance for these can be made in the plant design. Such information is particularly important since natural disasters, severe external impacts or other major catastrophes are all potential causes of a major accident involving loss of containment, cooling or radiation shielding.

Heat generation in storage is a function of the waste characteristics. The figure of about 17.5 watts (15 kcal per hour) per litre has been adopted as the design basis for some plants. Presently the tendency is to take safety margins and authorise a value of about 9 watts per litre corresponding to around 100 TBq/litre (2,500 Ci per litre). Although an extremely improbable event, loss of cooling over an extended time could in principle lead to a significant release of radioactivity to the environment. HLLW storage systems can however withstand loss of cooling for extended periods (of the order of a few hours) without significant risk of any activity release due to:

- the relatively low specific heat generation rate in the waste;

- the low normal operating temperature;
- the high thermal inertia of the system.

Containment failure is another potential cause of a large release of radioactivity through leakage. Containment failure can result from corrosion processes or accidental conditions.

Although the probability of fire to the supporting systems is relatively small, the consequences for the tank may be so severe that extreme care is taken to prevent it. Preventive measures include minimising the use of combustible materials and, the provision of early detection systems and extinguishers. Fire in electrical circuits is the most likely potential fire accident; it should be taken into account because it could result in a loss of control and a loss of power, leading possibly to a loss of cooling. Such an incident happened at the La Hague reprocessing plant and is reported on in section 7.3 of Chapter 7.

### *iii) Loss of cooling*

A serious accident would be caused by a prolonged loss of cooling in the high-level waste storage tanks if no engineered safety system was operating. It may be envisaged that, in such a case, the following sequence of events could happen:

- Phase 1: the temperature within the tank increases until it reaches the boiling point of the solution;
- Phase 2: the solution boils and eventually evaporates to dryness. During this phase, volatile and semi-volatile fission products, mainly ruthenium, might possibly be partly released through the off-gas treatment system of the plant to the atmosphere.

A number of studies have been undertaken in an attempt to quantify the time scale of semi-volatile radionuclide releases based on experimental (86) and engineering (87) models. Experimental tests have shown that, in the worst case of high-heat loading, 3 to 4 days are required before even a small (400 cm<sup>3</sup>) sample of HLLW solution evaporates to dryness and becomes a source of semi-volatile radionuclides (e.g. ruthenium-106) (169).

Preventive measures such as the redundancy of cooling systems and electricity supplies, and constant surveillance by man, provide the basis for a satisfactory level of safety. The parameters used in such studies must, wherever possible, be based on established practical data. Practical remedial measures could then be identified relevant to the time available for each stage of the sequence.

In reality, the probability of a loss of cooling situation lasting up to the completion of Phase 1 seems extremely low to all experts. Studies are being made however to further improve the reliability of cooling systems.

### *iv) Containment failure*

Apart from an extended loss of cooling, corrosion processes may be the main cause of containment failure in the high level waste tanks. Such processes have therefore to be taken into account at the design stage, by selecting materials and specifications as well as operating conditions likely to prevent or minimise their occurrence. Reference (86) gives a description of the various designs in use and the measures adopted to avoid significant corrosion effects. The main parameters are temperature, acidity, radioactivity, concentration and homogeneity of the waste solutions.

Corrosion can be accelerated by local heating, mechanical stress, addition of incompatible chemicals or construction errors, as well as other causes, and this could result in unexpected leaks.

Radiolysis may destroy nitric acid and increase the pH of the solution leading, in some cases, to precipitation of solutes. This can be detected through regular sampling and analysis and counteracted by the addition of nitric acid. Radiolysis will frequently accelerate chemical reactions and this effect should be anticipated. Sedimentation may cause hot spots and increase corrosion rates; sparging will prevent sedimentation. Normal venting arrangements give adequate protection against elevated or abnormally reduced pressures.

In storage tanks of modern design, containment failure resulting from corrosion is not likely to develop into the type of extreme situations sometimes envisaged for a complete and extended loss of cooling. A leak from a stainless-steel tank caused by corrosion would most probably be relatively limited in size initially and would increase only gradually with time. It would be identified at an early stage by the continuous monitoring system of the activity in the cell ventilation. Its spread would be limited if adequate secondary containment provisions are made. If the leak were in a cooling coil, it would be identified by the continuous monitoring system of the cooling circuit activity. Its spread would be limited by sealing the particular cooling circuit.

The tank contents are required, in case of emergency, to be transferable to a spare tank which is always available. In the hypothetical, very adverse, but exceedingly improbable situation where the additional barriers fail and are all defective to various degrees, the ion retention properties of soils around the tank might be such that most of the radioactivity released would remain confined in the immediate vicinity. Corrosion aspects therefore, while being extremely important, particularly from the point of view of operational reliability, are much less significant in terms of potential releases of radioactivity to the environment.

It has been assumed up to now that radiolysis of the aqueous waste solutions of highly concentrated fission products might generate enough hydrogen for a flammable mixture. Hydrogen could accumulate in the free space of the liquid storage tanks, and systems are usually provided for diluting this hydrogen by sweeping air over the liquor surface and venting it through the off-gas system. All forms of sparging tend to entrain droplets of waste solution, and this increases the load on the off-gas treatment system.

To investigate the possibility of radiolysis, laboratory experiments with real process solutions were performed in Germany for both dissolved fuel solutions and PUREX high level liquid waste solutions (burn-up 30,000 MWd/t: cooling time 420 days) (89). The  $G_{H_2}$  value (number of molecules  $H_2$  generated/100 eV of energy absorbed) was in the range of about  $1.5 \times 10^{-3}$  compared to the  $G_{H_2}$  value of 0.44 measured for pure water. Investigations carried out at WAK (90) have shown that the production and release of hydrogen by radiolysis in the HLLW amounts to  $1.5 \times 10^{-6}$  molecules  $H_2/100$  eV. This value is 1,000 times smaller than experimentally determined in laboratory conditions and  $3.10^5$  times smaller than the theoretical value of 0.44. Moreover, it was concluded that flushing of the gas above the HLLW to prevent the formation of explosive mixture is deemed prudent.

Transfer of HLLW from an old storage facility at WAK to a new one (LAVA) was carried out in 1987 at WAK (91). In addition to the conventional surveillance and monitoring equipment, the LAVA tanks have 13 thermocouples attached to the outer surface of this tank to detect the settling of high active solids on the walls. Each tank has a cooling capacity of 250 kW. An emergency system consisting of two diesel units, a deep well and other redundant electrical

equipment has been installed. About 63 m<sup>3</sup> of HLLW, which had been stored for 16 years, was transferred in batches of 1.5 m<sup>3</sup> by steam ejector. The amount of precipitate was found to be very small (~ 0.92 w/o) and the fissile plutonium content was only about 2.4 kg or 0.34 per cent of the total solids (700 kg). At the Eurochemic plant, about 47 m<sup>3</sup> of typical HLLW (called LEWC) was transferred to the vitrification plant PAMELA. No problem whatsoever was encountered with this transfer. The solid radioactivity deposited as a thin layer on the walls of the HLLW tanks is difficult to remove and its presence complicates the decommissioning operations.

The safety of high level liquid waste storage, as in all other steps of the fuel cycle, is based essentially on prevention rather than remedial actions. This is particularly evident from the design and construction of storage tanks and the emphasis placed on redundancy of systems, quality assurance and methods of surveillance. Strict quality assurance methods are applied to the construction of storage tanks. Nevertheless, prevention does not preclude the possibility of remedial actions which are often called for, and this possibility should not be neglected in safety assessments.

With regard to corrosion, laboratory tests under storage conditions and data derived from practical experience suggest that corrosion rates are very low, even for internal components which may be subject to more aggressive conditions than the tank itself. Most of the LWR-HLLW tanks are made of stainless steel (AISI 304 or 316 L) which have been shown to be very resistant against corrosion in HNO<sub>3</sub> medium.

Up to 310 tank-years experience of HLLW storage have been accumulated in France over the past 25 years (91). Over a ten year period, corrosion of 0.8 to 1.8 mg/dm<sup>2</sup> were measured in HLLW conditions. The influence of insoluble residues on the corrosion behaviour is important if hot spots are observed with local temperatures greater than 100°C. The FP concentration is equally important and should not exceed about 200 TBq per litre (5,000 Ci per litre). To avoid crystallisation of the HLLW solution evaporation below 150 litre per THM is not advocated. Concentration of HLLW from 5 m<sup>3</sup> per THM down to 500 to 300 litres per THM is acceptable.

The alarm instrumentation is generally duplicated and includes:

- high- and low-level alarms for the tank's contents;
- indication of activity in the cooling water;
- sump probe alarms;
- monitoring of the ventilation extract system and gas flow;
- high temperature alarms;
- tank surface temperature sensors.

#### 5.2.3.2 High level solid waste

##### *Waste solidification*

Storage of high level liquid waste in tanks is now generally considered an interim measure, and the conversion into stable solid form is considered as a necessary step for safe long-term storage and eventual disposal. The solidification process aims at transforming the aqueous solution of radionuclides into a chemically, thermally and radiolytically stable form. This process may also, in the future, be applied to some medium active liquid wastes (157).

Calcination can be a first step to reduce the mobility of the HLLW and to reduce its volume. The product remains partially soluble in water and is not suitable for long-term storage and disposal. The calcined radionuclides which are present in oxide form can be incorporated into a glassy matrix

by the addition of  $\text{SiO}_2$  and  $\text{B}_2\text{O}_3$  in appropriate ratios. Stable glassy borosilicate matrices can be produced by mixing 10 to 20 per cent radioactive oxides with 40 to 60 per cent  $\text{SiO}_2$  and 10 to 30 per cent  $\text{B}_2\text{O}_3$  and heating the mixture until about 1,100 to 1,150°C. A vitreous final form is chosen because it can be made safely and it offers good resistance to environmental influences, although it may devitrify at excessive ( $> 600^\circ\text{C}$ ) temperatures. According to the exact composition of the radioactive oxides other glass forming components, e.g.  $\text{Na}_2\text{O}$  and or  $\text{CaO}$ , are added. Addition of  $\text{Al}_2\text{O}_3$  and other oxides, influences the thermal stability and the melting point. HLLW can be directly added to the hot bath filled with molten borosilicates; this technique is based on heating the glass by its own electrical resistance and on the incorporation of a superficially formed oxide layer onto the glass body.

A second type of solidification process which is much less developed, consists of mixing mineral compounds (particularly those derived from  $\text{ZrO}_2$ ,  $\text{TiO}_2$  and  $\text{BaO}$ ) to form a crystalline insoluble matrix with the oxides from the radionuclides. The process occurs at high pressures and temperatures around 1,200°C. The resulting synthetic rock "synroc" is thermodynamically stable in geological strata.

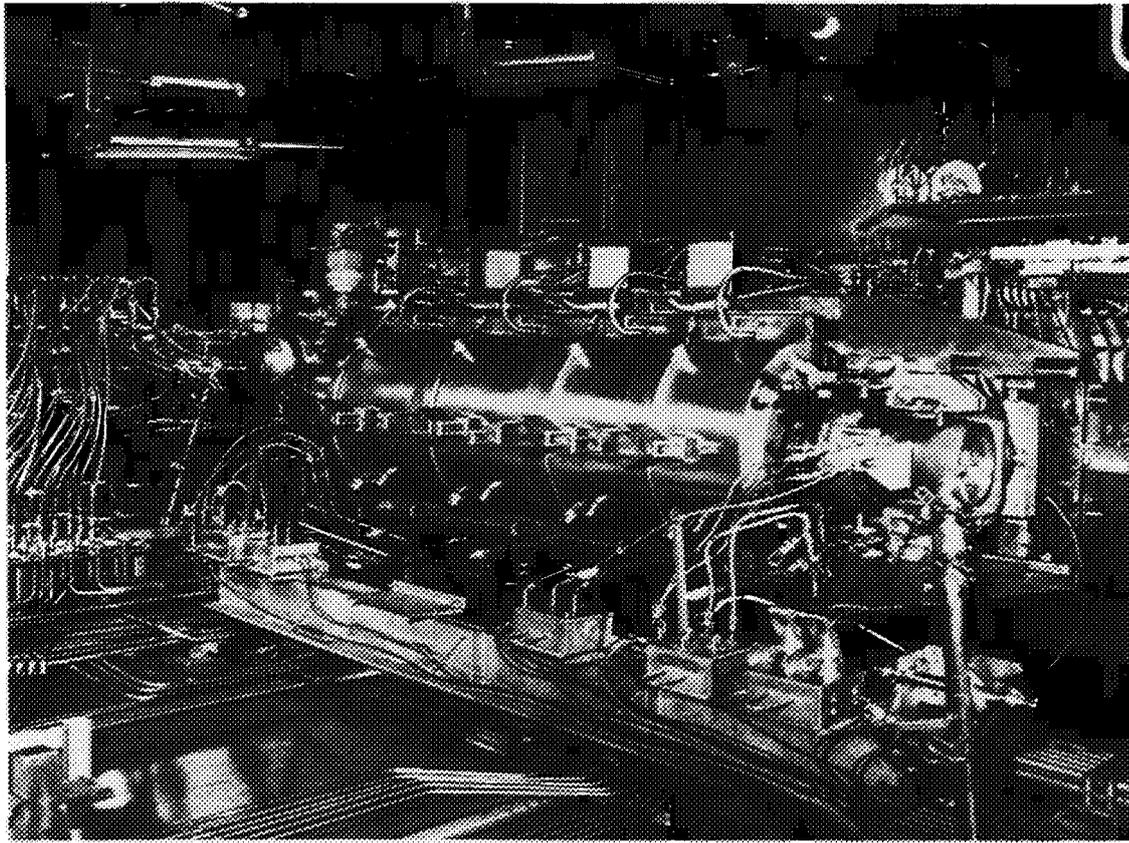
The solidification products from either vitrification or "synroc" process have to be enclosed in sealed metal containers for interim storage. The stainless-steel types are chosen for their resistance against hot molten slags or their use in hot isostatic compaction equipment but are not specifically selected for underground disposal. The use of overpack canisters should be considered.

#### *Technology and experience of vitrification*

Vitrification of HLLW is by far the most preferred solidification method. Two versions of this technique have been developed from bench scale to the industrial facility. The first vitrification process was introduced in France from 1958 and is based on calcination followed by vitrification of the calcined oxide. The first pilot plant PIVER was constructed in the 1960s and operated from 1967. A continuous vitrification process was subsequently developed at Marcoule and the AVM plant was constructed and operated from 1978 on; it has vitrified the whole stock of Marcoule's HLLW.

Early in 1980 a non-radioactive full scale prototype of the industrial vitrification facilities R7/T7 was constructed at Marcoule in order to check all technological aspects of the operation prior to active start-up of the R7 facility at La Hague (see Figure 5.16). Each of the industrial vitrification facilities has three lines each of 10 kg glass per hour. The glass volume in the canister varies from 100 to 120 litre/THM (275 to 330 kg/THM) (92) (See Figure 5.17).

The glass composition suitable for vitrification of the HLLW solutions from oxide fuel has been fixed at: 45%  $\text{SiO}_2$ , 14%  $\text{B}_2\text{O}_3$ , 10%  $\text{Na}_2\text{O}$ , 5%  $\text{Al}_2\text{O}_3$ , 12% fission products + minor actinides; the balance being made of  $\text{CaO}$ ,  $\text{Fe}_2\text{O}_3$ ,  $\text{ZnO}$ ,  $\text{Li}_2\text{O}$ ,  $\text{ZrO}_2$ ,  $\text{Cr}_2\text{O}_3$ ,  $\text{NiO}$  and  $\text{P}_2\text{O}_5$ . This glass has been submitted to chemical, physical and radioactive tests in order to determine its mechanical, physical, chemical and radioactive behaviour in representative conditions. The leach rate was determined on active samples and shown to be about 1 to 2  $10^{-7}$  g  $\text{cm}^{-2}$   $\text{d}^{-1}$  for the most significant radionuclides. This glass designated as "SON 68 18 17 L1 C2 A2 Z1" has been accepted by the safety authorities as an industrial reference composition and also accepted by the clients of the reprocessing plants.

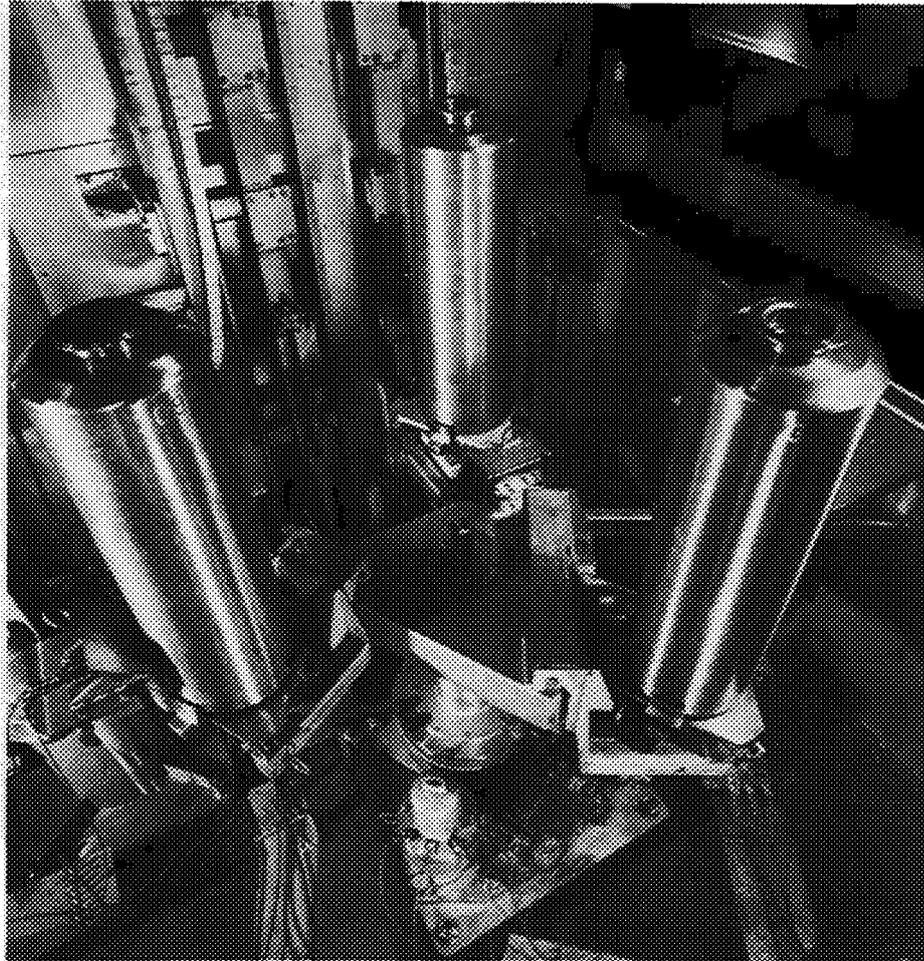


**Figure 5.16 COGEMA R7 vitrification plant in La Hague:  
inside view of one of the three main cells  
(calcination and vitrification cell)**

*(Source: Cogéma, La Hague – France)*

Recently (1989) the new industrial vitrification plant R7 was put into hot operation at La Hague. The vitrification facility (WVP) operating at Sellafield, adjacent to the THORP reprocessing plant, is based on the French technology but uses a slightly different glass formula (94).

An alternative direct vitrification technique was developed at KfK (Germany) in the 1970s and a pilot plant "PAMELA" was constructed in the period 1981-84 by DWK at the Mol-Dessel site in Belgium (95). The plant was put into active operation in 1984 and vitrified the HLLW forms of the former Eurochemic plant (96). This technique introduces the liquid radionuclides directly into the molten-glass bath without prior calcination. Up to now, about 460 tonnes of vitrified waste (about 2,100 canisters) have been produced by PAMELA (156). During the vitrification campaigns in the PAMELA plant a gradual enrichment in platinum metal concentration was observed on the bottom of the melter. A melter exchange was performed in 1986 without major problems. Similar demonstration projects have been set up in the United States at the West Valley plant (97) and will be constructed at Hanford. An industrial vitrification plant is under construction at Savannah River plant.



**Figure 5.17 Vitrified waste product containers on pouring station Carousel in the vitrification plant at Sellafield**

*(Source: BNFL Sellafield – The United Kingdom)*

PNC's research on vitrification technology began at Tokai, in 1975, with small-scale tests. Laboratory-scale active vitrification began at Tokai's Chemical Processing Facility (CPF) in 1982. Design work on the Tokai Vitrification Facility (TVF) began in 1980; the plant started cold-test operation in 1992.

The "synroc" process has been studied at the fundamental level and some bench scale work has been reported (98). The composition has to be adapted to the waste form and, since only defence waste has been considered, the results are hardly useful for typical acid HLLW solutions. However, the final product comprised 4 crystalline components; nepheline, perovskite, zirconolite and spinel, and has a principal advantage of being thermodynamically stable. The proposed technological concept includes, as a final step, hot pressing at ~ 1,100°C and at 70 to 140 bars pressure, which may pose problems in carrying it out with highly radioactive materials.

### *Safety problems*

The common feature of practically all high level waste solidification processes is that they require high temperatures and therefore extensive air-cleaning facilities for trapping the most volatile radionuclides which may be released from the feed solutions. These semi-volatile radionuclides potentially released during a high temperature process are: cesium, antimony and ruthenium. In order to retain these nuclides within the facility a series of air-cleaning techniques have to be installed. The most common are: dust scrubbers, high-temperature condensers, jet scrubbers or washing towers with appropriate scrubbing liquids, glass fiber prefilters and HEPA filters. Though the DFs in each of the air cleaning units are only moderate, the global effect is very substantial as DFs higher than  $10^6$  have been measured experimentally for each of the listed nuclides under carefully controlled conditions (99).

Overall DFs have been reported under routine conditions of hot demonstration facilities, e.g. PAMELA and AVM. The following data are given:

	PAMELA	AVM
Ruthenium-106	$1.3 \times 10^8$	$6 \times 10^7$
Antimony-125	$2.5 \times 10^8$	-
Cesium-137	-	$3 \times 10^8$

It is obvious from these results that vitrification operations do not generate a radiological impact on the environment provided sufficient care is taken to keep the individual filtration units in adequate operational order. Safety analysis studies show that the most significant accidents would be associated with the complexity of the off-gas cleaning systems required by the high temperatures involved (100). They could primarily result from errors in condenser operation. These accidents could lead to the release of semi-volatile fission products and have a radiological impact of the same order of magnitude as the annual dose commitment from routine discharges from the solidification plant.

Up to now no such accidental events have been reported during the cumulative periods of more than 30 work-years of hot vitrification plants in Belgium, France, Germany, the United States and the United Kingdom.

The risk of a serious mishap in the handling of the highly active feed solutions and in the operation of calciners and glass-melting furnaces is limited because the feed rate is relatively low (40 to 60 litres HLLW per hour). At such rates the release and spread of radioactivity from the feed solutions is confined to the process equipment or, in the case of a more significant incident, to the containment structures.

The most important safety problems which have been reported are related to the gradual contamination of the mechanical equipment, e.g. overhead crane, manipulators, etc. which has to be repaired or removed from the hot cells from time to time. In order to cope with these problems the new industrial vitrification plants (R7/T7 and WVP) are equipped with component dismantling cells, a very sophisticated remotely controlled handling system and fully automatic process equipment

where possible. After a first campaign in the PAMELA facility, a melting furnace has been remotely dismantled and replaced by a new one without major problems.

The filling of glass canisters by overflow casting or by underflow through a "freezing" valve does not have immediate safety implications but, since blockages of the "freezing" valve have been reported, it is important to equip the furnace with redundant outlets or with remotely operable outlets.

The composition of the active oxide has an influence on the overall behaviour of the melting furnace and on the quality of the glass. The platinum group metals and molybdates in particular have a very limited solubility in borosilicate glass (100). The addition of the insoluble residues from the post-dissolution feed clarification step to the glass is carefully controlled in order to avoid their local accumulation.

The fission product and actinide content of the glass has been reduced in France from 18 per cent initially to 12 per cent in order to reduce the heat output of the glass and consequently to lower the maximum center-line-temperature which may not exceed 620°C to avoid devitrification. The surface contamination of a glass container is severely limited in order to prevent spread of contamination.

#### *Prevention and remedial actions*

The usual actions have to be taken to prevent leakages from vessels and pipework and to limit their consequences if they happen. Suitable monitoring and process control instrumentation has to be provided for early detection of equipment failure.

#### *5.2.3.3 Fuel cladding waste, shearing and clarification residues*

Waste streams from current spent fuel reprocessing plants include, besides the high-active liquid waste, a number of highly active solid waste forms:

- fuel element cladding – either pieces of Magnox "swarf" from Magnox fuel, or "hulls" of stainless-steel or zircaloy from oxide fuels;
- some structural components from fuel assemblies, such as graphite sleeves, stainless steel grids and so on;
- solid residues, primarily left in the dissolver but which may also be found downstream, removed generally by centrifugation. These residues comprise small fragments of cladding and insoluble fission product and a small amount of fuel grains.

Magnox swarf is collected under water during the head end operations and is transferred, also under water, to storage facilities (see 5.2.4.2). Handled under such conditions it poses no fire hazard.

During the head end operations for oxide fuel, a quantity of solid waste material is produced as a result of the chop and leach process. The shearing operation produces small lengths of fuel encased in pieces of cladding but also some fine oxide and metallic powder from crushed fuel and cladding tubes. The cladding material of LWR fuel is made of zircaloy so the fine powders contain zirconium metal fines (101). Stainless-steel wastes pose no potential fire hazard and are not considered further in this section; their storage is covered in 5.2.4.2. The remainder of this section is primarily concerned with LWR-related wastes.

After dissolution, the basket is lifted from the dissolver or separated mechanically from the dissolver solution and washed to remove particulate material. The washed hulls are an important

type of waste material which is treated in a proper way to avoid uncontrolled release of radioactivity. The insoluble residues, together with some metal fines, are removed from the dissolver solution by centrifugation or filtration. Some fissile material, fission products and about 50 per cent of the tritium are attached to the hulls and "plated" on the inner surfaces of the leached hulls, with insoluble residues which contain essentially polymetallic inclusions of ruthenium, rhodium, palladium and technetium as well as chemically unstable fission products.

Between 3 to 4 kg per THM insoluble residues (102, 103) are produced in the reference LWR reprocessing head end. Due to the presence of ruthenium-106 and other relatively short-lived fission products these residues have a high-specific thermal output which varies between 300 W and 1 kW/kg according to the cooling time when very short-cooled fuel is reprocessed. They occur as a finely divided powder with a particle size of 0.1 to 50 microns depending on the time elapsed after dissolution and are mixed with zircaloy powder. The centrifuge bowls and/or the filtration equipment may need to be continuously cooled (particularly for FBR fuel) in order to avoid self-heating and potential pyrophoric reactions in the presence of air at high temperatures. However, the heat problem disappears as soon as the fuel is cooled sufficiently long, which is actually the case in reference LWR fuel reprocessing.

Once the filtered/centrifuged solids are extracted from the clarification rig they are stored under cooling or in oxygen free atmospheres. The potential pyrophoricity of zirconium and zircaloy fines associated with the insoluble residues is due to the chemical affinity of zirconium for O<sub>2</sub>. The pyrophoricity of zircaloy powders depends on many factors: the presence of oxygen, the temperature, the particle size, the "critical mass" and the dilution by other non-pyrophoric.

The insoluble residues collected in centrifuge bowls or other filtration rigs can be stored under cooling conditions. The storage conditions are carefully controlled and monitored to avoid overheating. Eventually the insoluble residues can be embedded in concrete for long-term storage or mixed with the HLLW to be vitrified. The long-term potential of these platinum group metals is best preserved by separate storage (106) although some research has already been undertaken into the feasibility of separating noble metals from residues (165, 166). When mixed with other glass-forming components in the vitrification process their mean concentration level in the glass and their dispersion in the glass block are carefully controlled in order to avoid their local accumulation (100).

The zircaloy hulls kept in the dissolver baskets are thoroughly washed, checked on their fissile material content and directly encapsulated (UP<sub>3</sub> situation) or transferred to a storage pool awaiting appropriate conditioning (UP2-400 situation) or encapsulation (THORP). The hulls have a low heat output, contain some fissile materials and are sufficiently gamma active to require remote handling in specially equipped facilities.

At present about 8,000 tes of hulls materials are stored or have been conditioned on line. The mean plutonium content is 23 g/te and the actinides caught inside the zircaloy cladding amount to about 14 g/te. The main fission products encountered are caesium-(137/134) and antimony-125, but these activities are negligible compared to the cobalt-60 activity resulting from neutron activation (185 to 555 TBq/te).

It may be concluded from these data that gamma scanning of the leached hulls does not provide any indication on the residual fissile material content; other methods, e.g. neutron interrogation, may have to be used. The tritium content of the zircaloy hulls is significant and ranges from 55 to 60 per cent of the total tritium production in the reactor. Any treatment by which the

hydrogen is expelled, or the hydride compound oxidized, will liberate this tritium inventory. Safety analysis of the conditioning facilities must take this fact into account in establishing the tritium release rates.

The conditioning of zircaloy cladding hulls has been investigated along different lines:

- incorporation in cement type materials;
- high-pressure compaction and incorporation in lead or graphite;
- melting with zircaloy (ingot formation);
- incorporating in a glass matrix.

Incorporation in cement type materials is the reference scenario which has been proposed in France and the United Kingdom as the industrial practice (160). It consists of conditioning freely packed zircaloy hulls in a Portland cement slurry kept in a 1.4 m<sup>3</sup> SS container (4,000 kg total weight).

A similar technique but preceded by partial rolling compaction leads to a reduced hulls volume, but the physicochemical and radiological phenomena are the same. The process was proposed in Germany and the products tested on representative scale with hulls from WAK. It appears that radiolysis of cement can lead to partial liberation of hydrogen and tritium. This phenomenon is assessed on its safety implications during the interim storage and the disposal operations.

Alternative techniques for hulls conditioning have been developed in Belgium, France and Germany (107, 108, 109, 110):

- High-pressure compaction and incorporation in low melting point alloys was studied and tested on bench scale with active hulls. The liquid metals dissolve the fission products and create a permeable matrix. The use of graphite-sulfur as embedment material for compressed hulls could provide a better homogeneity. This compaction-incorporation process has now evolved towards a compaction-encapsulation process. This type of process has the advantage of a better volume reduction (4 to 5) and absence of gaseous releases.
- At high-temperature (1,100°C) the melting process of (75% zircaloy - 25% copper) has been studied on active bench scale and provides a still higher volume reduction (5 to 7) but requires the manipulation of a relatively pyrophoric material at high temperature. The tritium inventory is almost entirely released during this process and therefore should be managed safely from the viewpoint of environmental protection.
- Incorporation of hulls in borosilicate glass has been tested on bench scale but does not lead to a reproducible and stable matrix.

#### 5.2.3.4 *Medium active and plutonium containing wastes*

Plutonium contaminated wastes are generated primarily during spent fuel reprocessing, plutonium conversion and MOX fuel fabrication. Each of these processes produces specific plutonium-contaminated wastes which have to be treated in a proper way, in order to transform them into a disposable form. In fuel reprocessing plutonium is accompanied by fission products and minor

actinides, whereas in plutonium conversion and MOX fuel fabrication americium-241 is the most important daughter-actinide which is formed by decay from plutonium-241 and which emits gamma rays.

The reprocessing operations produce a series of plutonium contaminated wastes which have to be treated separately:

- washing solutions from the spent solvent (Pu + FP);
- aqueous concentrates from the distillation steps;
- a variety of solid materials (combustible and non combustible).

The total loss of plutonium throughout the different high- and medium-level waste streams from 1 GWe year (27 te uranium-plutonium) ranges from 1 to 6 kg/Pu (111).

#### *Treatment and conditioning methods*

Waste solutions are either concentrated by evaporation and vitrified with HLLW, or treated chemically by co-precipitation and flocculation, dewatered and then stored or incorporated into bitumen (France) or cement (the United Kingdom).

About 18 m<sup>3</sup> bitumen is produced per GWe year and this quantity is packed into 100 canisters of 0.25 m<sup>3</sup> each (112). The most important risk in this process is a fire associated with the bitumenization step and particularly at the filling step of the drums (113). The fire risk arises in many ways, e.g.:

- temperatures exceeding 300°C are a direct fire risk;
- ignition by exothermic reactions with solvents, etc. may cause a fire.

Of all the known fires, each broke out in the drum, except in one instance when the overflow burned. The latency time of the overflow or fire varied, reaching up to 15 hours, with considerable smoke given off before the flames appear. One fire was reported in the Eurochemic plant at Mol-Dessel (114) in 1981 and was extinguished within a few minutes and without external contamination or radiation doses to the operators. Due to this incident, particular attention was given to the composition of the batches in view of exothermic reactions. Fire-fighting equipment and fire-detection systems have been installed at the new STE-3 plant of La Hague (113).

Another potential hazard with bitumenised waste is the radiolytic generation of hydrogen. Build up of gases in drums is prevented by *i*) provision of a sintered steel filter in the drum, *ii*) limiting the activity content per drum and *iii*) ventilation of the storage hall. In addition, air sampling of the storage hall atmosphere is undertaken routinely.

Solid plutonium contaminated waste arises at all stages of the processing operation, but particularly at the conversion of plutonium solutions into PuO<sub>2</sub> or UPuO<sub>2</sub>. The waste may present itself in different forms, e.g. contaminated paper tissues, rubber gloves, equipment, etc.

Various methods have been developed to treat plutonium contaminated solid waste to reduce their volume and/or to recover part of the plutonium present. These methods include compaction, incineration, acid digestion and acid leaching:

- Combustible waste with low plutonium content can be burnt in high-temperature incinerators, e.g. papers, solvents, organic materials. Several types of incinerator have been developed and are available and the safety problems associated with their operation are mainly situated at the off-gas purification stage and at the ash conditioning step (115). Corrosion of the gas purification equipment by HCl (and other) acids, and clogging of the filtration beds by airborne particulates, are the most important malfunctions which have been reported. The construction and maintenance of high-temperature incinerators is a complex task which needs a comprehensive study of the safety, maintenance and decommissioning operations at the design phase, in order to avoid mishaps once the incinerator is contaminated with plutonium.
- The plutonium content of various solid wastes can be reduced by washing. By using HNO<sub>3</sub> solutions, the plutonium content in the waste can be reduced substantially if the initial plutonium contamination was of soluble nature. The solutions can be reused after distillation thereby keeping the waste volumes low. If the solid wastes contain relatively large quantities of thermally treated UO<sub>2</sub>-PuO<sub>2</sub> residues, washing with HNO<sub>3</sub> at high temperature will be necessary to eliminate the plutonium content from the waste materials.
- Acid digestion (116, 117) has been thoroughly studied in order to recover quantities of plutonium from combustible waste at the EUROCHEMIC plant. About 600 kg of waste materials have been treated and 5 kg of plutonium was recovered which corresponds to a recovery factor of 97 to 98 per cent. At Hanford a similar facility was built and operated, but worked with lower initial plutonium concentrations. This process can treat plutonium from fuel fabrication plants and from reprocessing plants. However, the acid digestion process uses concentrated HNO<sub>3</sub> + H<sub>2</sub>SO<sub>4</sub> acids, which oxidizes the organic matter and dissolves the mineral materials, and requires temperatures of up to 250°C. The digestion step is followed by a liquid-liquid extraction step of the aqueous Pu(SO<sub>4</sub>)<sub>2</sub> solution. The safety problems encountered with this technique are numerous: use of highly corrosive acids at high temperature, deposition of Pu(SO<sub>4</sub>)<sub>2</sub> in the piping, criticality limits in the design and low throughput of the equipment. The technological feasibility has been demonstrated, but in the meantime the facilities have been decommissioned because of technical difficulties of maintaining the process equipment in a good condition over a long period of time.

Metallic waste contaminated with plutonium can be washed with acid, but electrochemical treatment has been shown to be a most effective process in reducing the residual contamination of the metallic pieces (glove boxes, process equipment, etc.). Normal chemical decontamination with acid (H<sub>2</sub>SO<sub>4</sub>) followed by alkaline-oxidizing (NaOH-KMnO<sub>4</sub>) treatments are very effective in reducing the surface contamination of metals but create a large volume of liquid waste to be treated.

*(continued on next page)*

*(continued ...)*

Electrochemical decontamination (also called electropolishing) has been tested on bench scale and demonstrated on the size of large nuclear equipment (118). Decontamination factors of  $10^4$  can be obtained in a short time and recycling of the decontamination electrolyte reduces the overall waste production to small-end volumes. This method is very promising in order to reduce the volume of technological wastes (119).

Technological waste originates in the reprocessing plant from equipment which is brought out and replaced by new items, and these contaminated pieces of equipment may be decontaminated, sometimes compacted, and embedded into cement type matrices (120, 121, 122). In principle these operations do not involve a special risk during their conditioning. However, if technological waste is heavily contaminated or contains significant quantities of organic materials and/or plutonium, gas formation and radiolysis might occur. The long-term structural behaviour of canisters filled with technological waste has to be examined closely as a preliminary assessment of the long-term storage safety and its behaviour in the repository.

A particular class of organic waste is the spent solvents resulting from the TBP used in the PUREX extraction or any other extractant used to purify plutonium. The large-scale use of TBP leads to significant volumes of spent solvent contaminated with plutonium, actinides and some fission products. However, by washing with alkaline salts or hydrazine containing solutions and following low-pressure distillation, radioactive contaminants and interfering degradation products of solvent are removed. Thus, the major fraction can be recycled to the process but a certain fraction remains unused and is treated as waste.

Different treatment methods for organic liquid wastes have been proposed:

- direct injection into an incinerator and combustion (115);
- pyrolysis in presence of  $\text{Ca(OH)}_2$  and combustion of butane (123 and 162);
- oxidative decomposition by peroxide treatment and neutralisation with  $\text{Ca(OH)}_2$  (124);
- direct treatment with  $\text{AlCl}_3$  and conditioning of Al phosphate (125);
- low-pressure distillation as reuse method of solvents (126).

#### 5.2.3.5 Gaseous waste management

Gaseous radionuclides are released when shearing spent fuel in the mechanical head end of the reprocessing plant and during the dissolution of fuel:

- tritium and carbon-14 are partially dissolved in the fuel and the cladding;
- krypton-85 is dissolved in the fuel and released during the head end operations;
- iodine is released during the fuel dissolution step.

Tritium, produced by ternary fission in the fuel, migrates towards the cladding and reacts with zirconium to form  $\text{ZrH}_2$ , a relatively thermally stable compound. Part of the tritium remains in the

fuel. During dissolution operations about 45 per cent of the tritium enters into the aqueous solution and follows the feed solution to the liquid extraction. The residual 55 per cent remains as  $ZrH_2$  in the hulls and follows the solid waste treatment process. Methods have been developed to reduce the tritium diffusion from the dissolver solution into the liquid extraction stages (127) and important technical developments were made in the 1980s, up to pilot-stage tests, on the removal of tritium from aqueous effluents (128, 129, 130). However, most reprocessing plants, being situated by the sea, do not envisage any tritium retention because the dilution capacity of the surrounding sea is sufficient to meet health physics standards and also the ALARA principle does not require its retention.

About 60 per cent of the carbon-14 is contained in the  $UO_2$  and released as  $CO_2$ , while 40 per cent remains inside the hulls because it is produced by neutron activation of nitrogen-14 which is present as an impurity in the zircaloy (131). The  $CO_2$  released from the dissolver can be trapped, as will be done on the THORP plant, by scrubbing with caustic soda and then precipitating out the carbonate by reaction with barium nitrate solution. It is furthermore advisable not to destroy the metallurgical structure of the hulls, e.g. by melting, in order to keep the carbon-14 inventory of the hulls unchanged. Therefore, careful assessment is necessary to select the best choice between encapsulation and release.

Iodine, and particularly iodine-129, is the only volatile radionuclide which is generally trapped in the reprocessing off-gas treatment rigs. It is released during the dissolution of fuel and trapped by alkaline scrubber liquids or silver-impregnated absorbents (136, 137, 150).

A lot of work has been done on the capture and immobilisation of krypton-85 from head end off-gases (133, 134, 135). Krypton as a gaseous fission product is released from the mechanical and chemical head end operations. Its radiological impact has been carefully assessed (136) and shown to be insignificant. Another reason for the decrease of interest in krypton capture methods was the technical complexity of the successive treatment steps (removal of moisture, oxygen and nitrogen oxides) requiring, inter alia, the use of hydrogen gas.

Finally the separation methods result in the formation of a very concentrated krypton fraction, absorbed on charcoal or zeolites, which has to be stored for a relatively long period of time ( $t_{1/2} = 10.5$  years) in order to decay. The storage of large amounts of krypton-85 might be considered as a greater potential radiological hazard locally, compared to the global impact of continuous dilution at the stack of the reprocessing plant. This situation should be reviewed if the world capacity of reprocessing plants increases drastically.

Application of the ALARA principle will assist any decision-making on the retention of krypton-85 and also carbon-14.

## 5.2.4 *On site interim storage of waste*

### 5.2.4.1 *Vitrified waste*

#### *Technology and experience*

Interim storage of vitrified waste is required for the period between solidification and transfer to a disposal facility. The period may last up to several decades, primarily to permit activity and heat generation to decay to a level acceptable for disposal.

Several concepts have been developed for interim storage, as described in the 1981 IAEA report (138), and they are: air-cooled vaults, water basins and shielded, sealed storage flasks. In all three cases the waste would be packed in standard stainless-steel containers. In the case of dry-vault storage, air cooling could be achieved either by natural or forced convection. Natural convection is, in principle, to be preferred since it is mechanically simple and reliable. However, if construction and size of the vault makes natural convection difficult, or if filters are to be used in the extract, then forced ventilation would have to be used.

Experience gained worldwide from the storage of irradiated fuel elements is of direct relevance to the water-cooled basin concept. In this case, forced circulation of the cooling water has to be employed and, to provide additional environmental protection, a secondary cooling loop is generally desirable. However, there are currently no facilities in use for storing vitrified waste by this means.

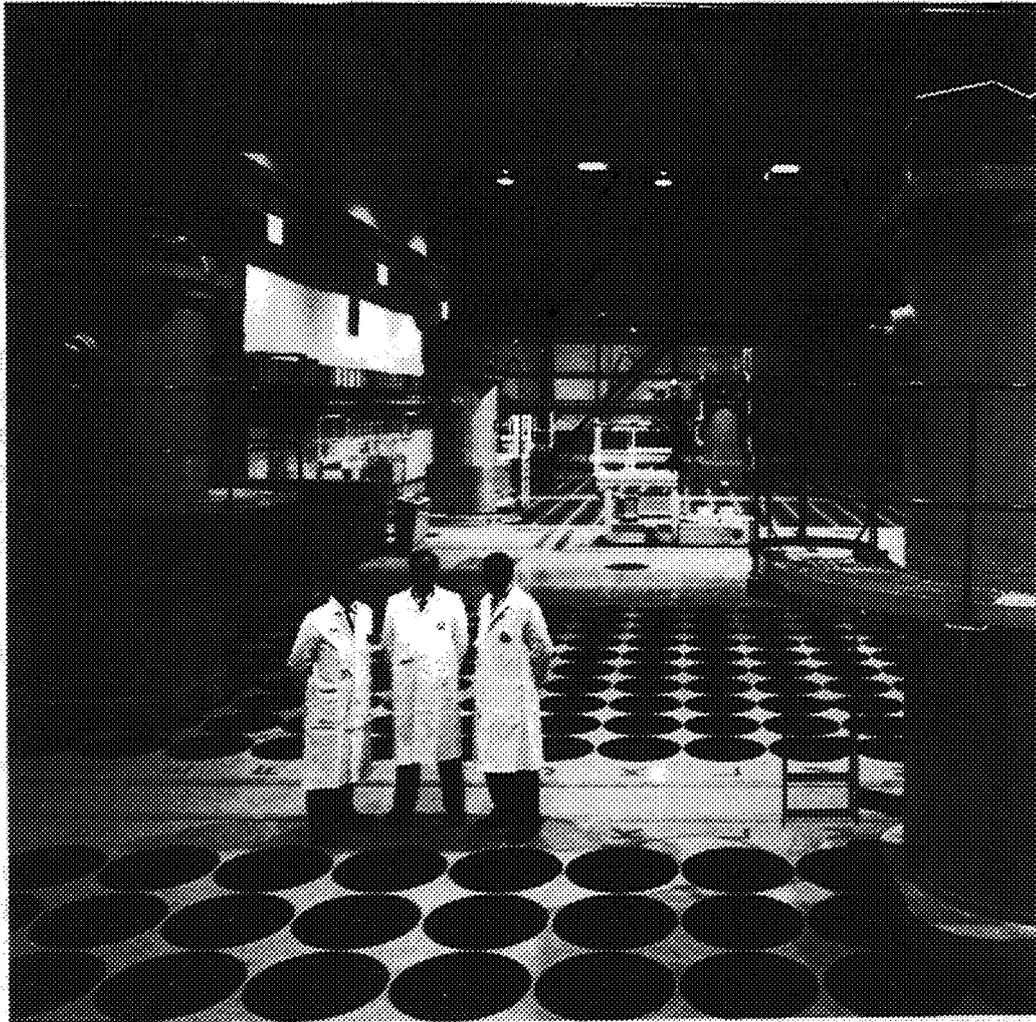
The choice between a dry vault or a water basin depends essentially on the total heat removal capacity and on the specific power output of each HLLW canister. Since the cooling period between the discharge of spent fuel and the vitrification of its HLLW has steadily increased, preference has been given to air-cooled vaults: these are operational in France, the United Kingdom, etc. and many are under construction (Figure 5.18).

The shielded, sealed-flask concept is inherently safe, requiring minimal surveillance since decay heat is dissipated by natural convection. However it has not been implemented commercially, though a few experiments have been conducted.

#### *Safety considerations*

As in the storage of spent fuel and of high level liquid waste, reliable and continuous cooling is important for safety to be assured. The consequence of a loss of cooling would, however, be less significant for vitrified waste because:

- the cooling time is longer;
- the waste form is a chemically and thermally inert solid;
- the storage system contains considerable thermal inertia which, after a long-term loss of forced cooling, permits natural convection to provide the main route for heat dissipation;
- the proportion of volatiles (notably certain fission products) is very low, as a direct consequence of the processes involved in forming the solidified waste.



**Figure 5.18 Vitrified waste storage**

*(Source: BNFL's Sellafield Plant – The United Kingdom)*

Nevertheless, the safety still depends principally on the reliability of the cooling system used – certainly in the initial stages of operation when the waste has a high rate of heat generation. As noted earlier most experience has been gained on air-cooled stores. In France, air-cooling is achieved initially by forced circulation and later by means of natural convection. The United Kingdom plant will rely upon natural convection from the outset, but forced draught fans are provided as a back up to cater for major reductions in convection due to circumstances such as adverse weather and any regulatory requirement to add filters to the extract. In such plants, containers are stacked in vertical shafts and the cooling air circulates up the shafts. The maximum glass temperature is generally maintained below 600°C, with outlet air temperatures between 100 to 150°C depending on process parameters. Stability of the concrete, or other structural materials are maintained at such temperatures over the lifetime of the store.

The design of vitrified waste stores must include allowances for external events such as earthquakes, floods and aircraft crashes. The probabilities of these are very site dependent. Bearing in mind the experiences gained through incorporating such allowances in the design of nuclear reactors and recent reprocessing plants, there should be no difficulties in the design of vitrified waste stores.

#### 5.2.4.2 *Cladding waste*

##### *Introduction*

Waste streams, besides the HLLW, include fuel element cladding, some structural components and solid residues. This waste generally generates less decay heat than the vitrified waste referred to in Section 5.2.4.1., so loss of cooling usually does not pose any major safety problems. A common feature of all these wastes is a very high radiation level due primarily to activation products, and this is mainly beta/gamma radiation. Actinide contamination (e.g. from undissolved, adhering fuel) can be as high as 1 per cent in some cases and this can give rise to significant alpha levels. However the main requirements are for heavy shielding and remotely controlled handling techniques.

##### *Magnox swarf*

Magnox swarf can be pyrophoric under favourable conditions (61); it can also react with water, to generate hydrogen, and this reaction may accelerate to such an extent that a hazardous "hydrogen excursion" may occur.

Storage of Magnox swarf from the early nuclear programme, particularly in the United Kingdom, was in dry concrete silos. No pyrophoric events have been recorded in such silos; nevertheless those dry silos at Sellafield (the United Kingdom) dating from the 1950s have recently been equipped with a rapid-response argon inerting system to bring the fire prevention up to modern standard.

There has been one reported fire in a dry storage silo, this occurring in 1981 at La Hague in France. Two bunkers, each with a volume of about 2,500 m<sup>3</sup>, were designed for storing graphite sleeves and magnesium end fittings from graphite-moderated, gas-cooled reactor fuel elements. A slow-burning of some of the contents of one bunker was initially detected by radiological surveillance. Nitrogen purging, followed by the injection of large quantities of water, succeeded in controlling the situation. The presence of fragments of uranium combined with mechanical shocks during the last pouring of the waste into the silo was thought to cause ignition of the fire. Since the incident, the equipment and building have been refurbished and additional precautions have been taken to prevent similar events.

The majority of Magnox swarf, however, is stored under water in concrete silos. Generation of hydrogen is normally slow but the reaction also generates heat and, if the heat transfer away from the silo is poor, it may result in a thermal runaway reaction ("hydrogen excursion") and a very rapid evolution of hydrogen. The heat generated may lead to structural damage of the concrete silo, and hydrogen concentration above the water level can reach 40 to 50 per cent (inside the detonation region). Cooling coils are often incorporated in silos, but the swarf itself may restrict convection currents, retarding the cooling, and "excursions" do occur. Many of the early wet silos probably no longer contain Magnox swarf – most has reacted with the water to form dense sludges. More recent silos are equipped with argon/nitrogen inerting systems, activated by temperature and/or hydrogen sensors in the silos, which are intended to control the oxygen concentration to below flammable limit should an "excursion" occur.

On account of the problems of storing Magnox swarf in either dry or wet silos, such storage is regarded as an interim measure whilst the swarf awaits encapsulation or immobilisation prior to disposal (see Section 5.2.3.3).

### *Stainless-steel waste*

This waste may include stainless-steel hulls from oxide fuels, and structural supports (grids, spacers, etc.) from fuel assemblies. This waste poses very few problems, other than high-radiation levels, and is generally stored in silos. Occasionally this waste is stored together with zircaloy hulls, but the safety aspects are not associated with the steel components. Nevertheless this is only an interim storage method. In France, they now encapsulate the steel and components with zircaloy hulls in concrete for disposal. R&D work in France is investigating volume reduction of hulls by melting and compaction (139).

### *Zircaloy hulls and end caps*

In the UP2-400 plant at La Hague, zircaloy hulls and end caps coming from shearing and dissolution of the LWR fuel were transferred in bulk into a wet silo until 1989. Hulls and end caps were transferred from the shearing cell to the silo inside a hydraulically driven drum through a sloping pipe which ends in a cell located above the silo. The drum is emptied in this cell and comes back to the shearing cell. The concrete silo has an inside stainless-steel liner. Silo water is maintained at a pH near 4. The silo plenum is continuously vented.

Although no safety problems arose from operating the silo, the French Safety authority asked COGEMA for a more readily retrievable storage. A new organised storage was provided in the ponds of the former gas graphite fuel decanning plant. The drums are transferred from the cell to the pond by a cask and stored under water. Hulls and end caps are put inside separate drums. Pond water has the same treatment as fuel storage pond water. Its activity upper limit is 185 MBq/m<sup>3</sup>.

In the UP3 plant and in the future UP2-800 plant, hulls and end caps are loaded on-line in stainless-steel containers and embedded in concrete. The containers are stored in a concrete building (EDS).

### *Graphite waste*

The head end process for AGR fuel in the United Kingdom generates a substantial volume of active graphite, from removing and crushing the sleeves of each fuel assembly. This is presently stored in steel containers in air-cooled stores (similar in concept to those stores for vitrified waste). The waste is initially wet, from the process, and hydrogen evolution by radiolysis necessitates filtered venting of the steel containers. This waste is destined for eventual encapsulation and disposal.

#### *5.2.4.3 Insoluble residues (see also 5.2.3.3)*

Residues from spent fuel dissolution consist mainly of insoluble noble metals, ruthenium, rhodium, palladium, molybdenum and zircaloy or stainless-steel fines produced in the shearing of the fuel. Typical arisings are 5 kg/THM fuel; they have a high specific heat output of about 300 W/kg due primarily to short-lived ruthenium-103 and rhodium-106 fission products, and may contain small amounts of undissolved uranium and plutonium.

The storage of residues, like hulls and other fuel wastes, is temporary. Safety can be improved by keeping the residues in a form which is less dispersible. Work over the past few years has suggested that the residues can be incorporated into vitrified waste, and France and Japan have taken up this option. In the United Kingdom, residues and fines may be incorporated into concrete after appropriate cooling.

In France, dissolution fines mainly arise from the operation of the centrifuge used to clarify the solution of spent fuel dissolution. In the UP2-400 plant, the dissolution fines are stored in the hulls and end caps silo. Fire protection equipment is required. In the UP3 plant and the future UP2-800 plant, the dissolution fines are stored under water in tanks very similar to the HLLW tanks. In particular, reliable stirring and cooling systems are provided to prevent accumulation or drying out of fine particles. The dissolution fines are going to be vitrified together with the HLLW.

The majority of these residues remain inside the dissolver from which they are periodically flushed out and separated by decantation. Some fine residues are carried over in the dissolver liquors and are removed by centrifuging. Whilst the latter residues can be retained and stored in the centrifuge bowls, generally they are flushed out and added to the coarser residues in storage tanks.

### 5.3 Decommissioning of nuclear facilities

#### 5.3.1 Decommissioning philosophy

As in the case of conventional industrial plants, much can be done to safely extend the normal lifetime of nuclear plant. There is nevertheless a limit to the cost-effectiveness of maintenance and improvements/modifications, and a point will come when it will be no longer economic or safe to keep the plant running. Environmental standards now demand that shutdown plant or sites be rendered/maintained safe and left in a visually acceptable state. Given the problems of residual radioactivity in nuclear plants, it is especially important that adequate steps are taken to avert possible environmental damage. Interest in the treatment of redundant nuclear plant has been increasing steadily over the past decade. A responsible nuclear industry is one which can demonstrate its ability to deal with the end-product in a publicly acceptable manner; this is the basis of the large body of work being carried out under the general heading of "decommissioning".

The term "decommissioning" usually refers specifically to the steps to be taken when a nuclear facility has ceased operation completely, without intent to restart at any stage. It is conventionally described for reactors as a three-stage process. However, as decommissioning of fuel cycle facilities has proceeded in the past decade, it has become clear that such facilities cannot readily be considered under the three-stage classification. This is because fuel cycle facilities differ from reactors in:

- radiological condition – loose (and therefore potentially easily removed) contamination as opposed to "locked in" activity, and with a wider range of activity levels;
- structural aspects – the most hazardous materials are often in a fairly basic primary containment rather than huge concrete structures;
- process aspects – an integrated reprocessing plant should already have waste treatment and handling facilities which could deal with decommissioning wastes.

For fuel cycle facilities (140) the following decommissioning stages may be more appropriately considered:

*Initial decommissioning.* This stage usually quickly follows "Post Operational Clean Out"; this status is one in which safety is assured by minimum surveillance and maintenance. It is broadly equivalent to the reactor Stage 2 status.

*Dismantling.* This is the most expensive part of the process where the plant and equipment, which were in contact with radioactive materials, are removed, treated and

packaged. At the end of it only low-active or trace-active cells or buildings are left. The magnitude and cost of dismantling depends, to a large extent, on the criteria for the level of residual activity which will be tolerated after decommissioning. The so-called "de minimis" criteria will have to be very well-established to improve the planning, and to reduce the volumes of radioactive waste.

*Demolition.* The removal of residual structures is referred to as demolition.

The process of decommissioning any plant or structure requires preplanning to a degree consistent with the technical complexity, potential environmental impact and regulatory requirements. Features to be incorporated in a decommissioning plan have been described in many sources (141, 142, 143). These include:

- records of all operations within the facility, including incidents;
- plant description prior to decommissioning – includes up-to-date drawings, historical recollections, old drawings, lists of modifications;
- description of decommissioning activities;
- technology requirements – available techniques, and those requiring further development;
- estimates of waste generation – types, quantities, options for treatment, storage and/or disposal, "de minimis" criteria (if any);
- safety analysis, including environmental impact assessments; of decommissioning activities in view of the state of plant to be reached on completion of decommissioning;
- description of the organisations involved – including QA and training requirements;
- radiological protection measures – including ALARA. Experience has shown this is an important consideration;
- description of the surveillance and maintenance programme needed at the end of the decommissioning stage – including balance of costs, doses to operators and collective dose of workforce during the campaign.

Some well-presented examples can be found in (143).

Of equal importance to planning is the programming of the various decommissioning stages. Much of the contamination in fuel cycle facilities may be removable, and its removal is often highly desirable in order to simplify the technology required for dismantling and to reduce the potential waste categories. Whilst activity levels generally decrease with time, plants containing plutonium (and uranium-232), will show increases in activity over a few decades due to in-growth of americium (or other nuclides, such as thallium-208). The timing of decommissioning stages must therefore take account the type and amount of nuclides present. In general, fuel cycle facilities fall into one of four classifications, according to their radioactivity:

*Plutonium.* No radioactive decay but detrimental in growth of americium; potential risk to operators and the environment if decommissioning is delayed.

*Low-beta/low-gamma.* Activity levels manageable even in early stages of decommissioning with low technology requirement. With a mixture of nuclides, little benefit results from delay of decommissioning.

*Medium-beta/medium-gamma.* Some decay benefits may be obtained from deferring decommissioning for 30 to 50 years; low technology requirement.

*High-beta/high-gamma.* Intense radiation, dominated by long-life fission products; no deferral advantages, and high-technology requirements (remotely controlled handling).

### 5.3.2 Decommissioning technology

It is not the purpose of this report to describe the multitude of techniques available, and under development, for accomplishing decommissioning work. A number of conferences have been held in recent years on the topic of decommissioning techniques (141, 142, 143), where they are discussed in detail. Such conferences also address the handling and treatment of decommissioning wastes.

The Commission of European Communities organises five-year development programmes on decommissioning, and the 1984-88 programme included a substantial proportion of work on fuel cycle facilities (144). The Nuclear Energy Agency's Collaborative Venture promotes exchanges between the Member countries. The more recent decommissioning projects supported by CEC and/or NEA comprise (140):

#### CEC research programme

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United Kingdom	BNFL	Mixed Oxide Plant
France	CEA	PIVER Vitrification Plant Fuel Fabrication Complex
France	COGEMA	AT.1 FBR Pilot Reprocessing Plant
Belgium	Belgonucleaire	Mixed Oxide Plant
Germany	NUKEM	U-Th Fabrication Plant
Italy	ENEA	Pilot Reprocessing Plant
Japan	RANDEC, PNC	Nuclear facilities

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#### NEA collaborative ventures

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United Kingdom	BNFL	Mixed Oxide Fuel Plant
France	CEA	FBR Pilot Reprocessing Plant
United States	DOE	West Valley Reprocessing Plant
Belgium	Belgoprocess	Eurochemic Reprocessing Plant

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### 5.3.3 Radiological protection aspects of decommissioning

The formulation of detailed decommissioning plans requires the establishment in some detail of activity levels pertaining to the plant to be decommissioned. The assessment of the activity present is to assist in defining health physics requirements and in estimating the volumes and types of waste to be produced.

Assessment of the activity in fuel cycle facilities is not easy; the contamination may not be fixed, and isotopic distribution will vary. Health physics records, extending back over the entire life of the plant, are likely to be incomplete. Generally, therefore, a health physics survey is an essential prerequisite to decommissioning a facility. The objectives of such a survey are to optimise:

- the man hours involved in each task;
- the dose budget (i.e. the radiation exposure of the personnel involved in the work);
- the quantities of radioactive waste (taking into account disposal requirements and routes);
- the monitoring procedures for the decommissioning operations.

The survey itself could include the following activities:

- measurement of dose rates in areas where personnel will be working;
- measurement of surface contamination in those areas;
- taking samples of contaminated structural or containment materials to determine the likely amount and activity of radioactive wastes which will be produced. This could be reduced by further decontamination, but it should provide an upper-limit estimate.

As a general rule, decommissioning should involve the following operations which have radiological protection aspects:

- removal of all major sources of activity, fuel pieces, solutions, sources, using either existing routes or "ad hoc" means;
- evaluation of the radioactive inventory remaining (principally contamination);
- preparation of detailed decommissioning plans;
- remote decontamination of external surfaces and decontamination of vessel and pipework interiors by prolonged steeping or flushing with chemical reagents;
- remote handling and/or cutting of contaminated equipment or material;
- controlled man entry for further decontamination and breaking down of large items;
- dismantling of the facility;
- management and disposal of the wastes generated;
- measurement of the residual radioactivity in the building and on the site;
- control of the released material/equipment.

Provided that an adequate health physics survey was conducted and the plant protection features (including air extract and cleaning facilities, containment barriers and waste management facilities) can be kept operating – or replaced by temporary equipment, at least during the initial stages – there is no reason why there should be an unacceptable risk of an incident with significant radiological consequences for the environment. The risk ought to be no greater than that arising during normal operation of the plant and will decrease as the radioactive inventory decreases.

### 5.3.4 *Decommissioning of front end fuel cycle facilities*

Some decommissioning of front end facilities has been undertaken during the last few years, but very little information has been published. The decommissioning of the gas-diffusion enrichment plant at Capenhurst, the United Kingdom, has completed the high-enrichment end of the plant and is now progressing towards the lower enrichment stages. Decommissioning of seven facilities at AWE Aldermaston and Harwell included plants which are comparable to front end plants, such as uranium workshops and a rolling mill (143).

### 5.3.5 *Decommissioning of back end fuel cycle facilities*

#### *Status*

In recent years, there has been a number of such facilities (or parts of facilities) cleaned out, or even fully decommissioned. In the United Kingdom several small-scale installations or rigs were completely dismantled many years ago (145) but, more recently, BNFL have been undertaking pilot decommissioning work on a disused mixed oxide (co-precipitation) plant at Sellafield where the major active material is plutonium (143). This has been used to test a variety of techniques – in situ assaying, solvent cleaning of components, temporary containments, decontamination techniques and size reduction. The use of strippable coatings inside temporary containments is reported as successful in reducing contamination levels and dose uptake. One major aspect of the BNFL work was the decommissioning of a number of alpha-contaminated glove boxes.

Some good details of the successful decommissioning of glove boxes in the Belgonucleaire mixed oxide fuel fabrication plant, with particular reference to the techniques used in the work, are in (144). Some of the glove boxes were substantial (up to 10 m<sup>3</sup>), with residual plutonium contents typically 100 to 200 grams. This work was partly complemented by an EEC Research and Development trial (144).

Vessels and evaporators at Tokai Reprocessing Plant and glove boxes in the MOX Fabrication Plant have been dismantled in Japan, with the main plants themselves still in operation. A systematic programme has been initiated, for the development of decommissioning technology, using JAERI's reprocessing test facility.

At Dounreay, Scotland, the UKAEA successfully decommissioned and reconstructed a substantial FBR irradiated fuel reprocessing plant between 1973 and 1980 (11). Nearly all sections of the original plant had to be entered, after decontamination from levels as high as 100 R/hour and 1,000 cps alpha. A number of techniques were developed to assist in the decontamination.

The Eurochemic reprocessing plant in Belgium was shut down in 1974. Since then a number of refurbishment options were considered until 1986 when the last of such projects was abandoned – primarily because of lack of financial interest. A number of studies were then carried out on decommissioning options (143) and in late 1987 funds were granted to decommission some areas (extraction building, analytical laboratories, HA liquid project) as well as uranyl nitrate and PuO<sub>2</sub> storage buildings. All process equipment was removed, and the floors, walls and ceilings of the buildings were decontaminated to background level. The buildings were then withdrawn from the radiologically controlled area, demolished and subsequently disposed of as industrial waste on a public tip in December 1990. Decommissioning of analytical laboratories and the main extraction building has since commenced.

As part of the EEC's R&D programme, decontamination and dismantling of the PIVER plant was carried out between 1986 and 1989 (144). This pilot vitrification plant at Marcoule, France, had operated from 1969 to 1980, and the aim was to eventually re-use the cell for a new vitrification procedure. The R&D work involved multi-stage decontamination and cleaning of the cell and its equipment, followed by remote dismantling of the internals.

In the United States during the late 1980s, a major decommissioning task was undertaken at the Strontium Semi works Pilot Fuel Reprocessing Plant at Hanford. The plant comprised 11 structures, including the main process building and a 61 m stack. Unlike the examples noted previously, concrete "entombment" was an option seriously considered for this project and the final decommissioning plan reflected a combination of dismantling and entombment techniques for the buildings. An earthen barrier, 4.6 m thick, was then placed over the entombed structures. The dose commitment for the partial dismantling/entombment option was, at 12 to 21 man rem, only 10 per cent of that for the total dismantling option.

Much of the decommissioning of nuclear fuel cycle facilities has been conducted by the relevant utilities or licensees. There is scope for more involvement of the private sector, however, and decommissioning of laboratories at UKAEA Harwell and CEGB Berkeley, in the United Kingdom, have illustrated both the advantages and pitfalls of such an approach (142, 143, 145). Although not specifically fuel cycle facilities, the lessons learned are still relevant to this section; in particular the provision, by the contractor, of his own services and facilities to support the work, and the contractor's lack of "local knowledge".

#### *Safety analysis*

As noted in Section 5.3.1, radiological conditions in fuel-cycle plant vary, with potentially substantial amounts of loose or easily mobilised active materials spread through much of the equipment and cells. The overriding safety considerations are aimed at keeping worker doses as low as reasonably achievable (146). This can be assisted by:

- choosing techniques which minimise cross-contamination;
- minimisation of wastes, especially secondary waste;
- provision of adequate, temporary, containment and ventilation during decommissioning;
- provision of temporary shielding;
- choosing appropriate working practices to contain radioactive wastes, segregate them and pack them for disposal;
- measurement and control of worker doses (external and internal);
- adequate training of workers (use of computers, videos and mock-ups).

Such aspects should be incorporated into a well-prepared decommissioning plan (section 5.3.1) which is an essential precursor to safe decommissioning.

#### *Operational experience*

In summary the experiences have been successful and without any severe/major incidents, but a number of safety-related lessons have also been learned to date, many of them relating to philosophies of design and operation of facilities which gave little thought to decommissioning. Some of the more important ones are as follows:

- A well-prepared decommissioning plan is essential.

- It is important to know the initial radiological conditions existing in the facility prior to decommissioning. Lack of knowledge can lead, for example, to additional work and dose uptake.
- The application of the "de minimis" principle (if any) to a decommissioning project is important in the planning stages.
- A complete set of drawings for operating plant should always be kept up to date to reflect plant modifications.
- A record of all operations within the facility should be maintained; this should include all incidents (e.g. liquor spills) and subsequent attempts at decontamination.
- When work in a facility ceases, some planning for decommissioning ought to start immediately whilst staff with a knowledge of the facility are available.
- During design of new facilities, much more attention should be paid to building in features which will aid decommissioning (e.g. modular design, easily decontaminated equipment and surfaces).

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## 6. Transport of Radioactive Materials

The OECD Member countries with a nuclear programme have more than 30 years of experience in the transport of radioactive materials and a great deal of statistical data already exists on the subject.

Specific safety regulations are well-developed and harmonised internationally since they are all based on the IAEA recommendations (1, 2, 3). Accident situations have been taken into account from the beginning and the transport regulations are framed so that unacceptable radiation exposures should not occur from radioactivity release even in the case of severe accidents. Test procedures verify that actual designs comply with these regulations and Table 6.1 cites these for the main types of packages.

Table 6.1 Main tests for packages

Type A package	<ol style="list-style-type: none"><li>1. Free drop test = 1.2 m</li><li>2. Water spray test</li><li>3. Compression test</li><li>4. Penetration test</li></ol>
Type B package	<ol style="list-style-type: none"><li>1. Drop of 9 m onto an unyielding target</li><li>2. Drop of 1 m onto a bar end</li><li>3. Thermal test at 800°C for 30 minutes</li><li>4. Water immersion test under 15 m</li></ol>

The experiences acquired over the years were reviewed thoroughly during the Tenth International Symposium on the Packing and Transport of Radioactive Materials which took place in Yokohama, Japan in September 1992. As far as is known, no accidents have ever been encountered with serious radiological consequences.

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Test specifications for type B packages are related to potential accident situations and the package is required to be so designed that, if it were subjected to these tests, the integrity of the containment of the radioactive contents and eventually of the shielding would be retained to the extent specified in the IAEA regulations, in which the nature of the contents is taken into account.

The safety tests on type A packages are based on the premise that the package may be destroyed in a severe accident but require that it withstands normal transport conditions and foreseeable accidents such as a drop in handling, whereas the regulations define the type B package as being able to withstand severe tests corresponding to accident conditions in transport with reasonable safety margins. Free drop impact and fire are of prime concern for type B package tests:

- The drop test corresponds at least to the stresses sustained by the package in the collision of the shipping vehicle on a bridge pile or a concrete wall at a speed of 100 km per hour. The validity of this test has been endorsed by several accidents, such as a drop of enriched UF<sub>6</sub> containers from a carriage onto the track at a speed of 100 km (Germany, 1972 and the United States, 1977), the drop of a spent fuel flask from a highway bridge (the United States, 12 December 1970), and from a crane (Belgium, 1973). Sandia Laboratories at Albuquerque, New Mexico (the United States) carried out in 1977, full-size accident tests involving spent fuel flasks, and truck and locomotive collisions at speeds higher than 100 km per hour (5). In the United Kingdom in 1984, a full-size test involved a train at speed of 160 km per hour crashing into a spent fuel element canister. The test showed that the canister could withstand such an impact (6).
- The fire or thermal test has been specified on the basis of the statistics of airplane and oil tank lorry fires. These conditions are unusual in practice and very difficult to achieve experimentally. Present insulation materials allow packages to withstand fires more severe than those specified in the test regulations.

The transport of fissile material requires consideration of any problems which might arise in the event of a criticality accident. The IAEA regulations require that fissile material packages have criticality approval and this approval must take into account the results of both the various tests to which the packages have been subjected and the accident assessments. A mechanical impact which could bring into close proximity several nuclearly safe quantities of fissile materials by deforming the packages typifies the remote accident situations which have to be considered for type B packages in the criticality safety clearance context.

Practical accident scenarios severe enough to damage the containment and give rise to contamination of the environment are difficult to conceive for adequately designed and maintained packages. Contamination levels would depend on the nature and radioactive inventory of the contents and the possibilities for operator intervention (7, 8). (No environmental contamination resulted from any of the fuel cycle transport accidents listed in Table 6.3)

The possibility of a ship carrying radioactive materials accidentally set on fire or sinking after a collision or another failure has also been considered. Risks from fire and explosion on board of the vessel can be mitigated by adequate protective measures. Even if a ship were to sink, most packages would survive unless the depth exceeds 200 m, however, for the most hazardous materials, packages would survive in much deeper water. The salvage of the packages in the Mont Louis incident shows that, in relatively shallow waters, packages can be recovered virtually undamaged (see 6.5).

## List of abbreviations

<b>AFR</b>	Away From Reactor
<b>ALARA</b>	As Low As Reasonably Achievable
<b>AR</b>	At Reactor
<b>BWR</b>	Boiling Water Reactor
<b>CANDU</b>	Canada Deuterium Uranium
<b>DBE</b>	Design Basis Earthquake
<b>DIN</b>	Deutsche Industrie Norm
<b>DWL</b>	Demand Working Level
<b>FBR</b>	Fast Breeder Reactor
<b>FP</b>	Fission Products
<b>GCR</b>	Gas Cooled Reactor
<b>HAZOPS</b>	Hazard and Operability Studies
<b>HLLW</b>	High Level Liquid Waste
<b>HTGR</b>	High Temperature Gas Reactor
<b>HAZOPS</b>	Hazard and Operability Studies
<b>IAEA</b>	International Atomic Energy Agency
<b>ICRP</b>	International Commission on Radiological Protection
<b>IDR</b>	Integrated Dry Route
<b>KfK</b>	Kernforschungsanlage Karlsruhe
<b>LWR</b>	Light Water Reactor
<b>MOX</b>	Mixed Oxide

### *Heat transfer*

In dry transport, heat is mainly transferred by radiation (~ 10% only by convection). The cladding temperature essentially depends upon surface state and emissivity level. 400°C is typical for current designs. In wet transport the cladding temperatures are usually kept below 150°C by either natural or forced convection cooling; the former is, however, preferred in the majority of cases. Heat is transferred to the flask surroundings by natural convection and radiation.

Sometimes materials which are included in flask designs for radiation shielding also serve for fire protection purposes. Shielding consisting of wet plaster or wet, aluminous cement layers is of this type. If the flask is involved in a fire, the wet material is dehydrated and an efficient insulating layer is formed.

### *Radiation shielding*

Lead is often used in spite of its unfavourably low-melting point (327°C). Cast iron or steel are also used, but their use can lead to high weight and large dimensions. The flask opening is usually closed by a barrel or sliding door. Further shielding is provided by the water (in wet flasks), resin or polyethylene.

### *Mechanical design*

The main steel-flask envelope provides the necessary resistance to impact and puncture. Shock-absorber devices are placed at each end of the flask to reduce the consequences of impact. Wash-out plugs are protected by covers.

Transport of spent fuel elements in canisters by truck or by railway wagon is presently the most widespread industrial practice within the continental zones (the United States, Europe) (14). International and national regulations are worked out in detail to prevent radiation hazard in case of incident or accident (1).

Increased attention has recently been given to transport by ship or barge in the United Kingdom, Canada, Sweden etc., as a safer and less-costly alternative to road and rail transport. Ships owned and operated by nuclear fuel cycle organisations or nuclear power plants provide increased safety since they are specially built or modified for these types of transport. Trained personnel present on the ships is an additional safety asset (14, 15, 16).

Safety studies on abnormal events occurring during ocean transport have been carried out in Japan (17).

## **6.4 Transport of vitrified high level waste**

The problems encountered with the transport of vitrified HLW are very similar to those of spent fuel elements: high gamma radiation and heat output. Though the total radioactivity of vitrified HLW canisters is very high (up to 17 000 TBq/canister) the mean gamma energy is lower than that of an equivalent amount of spent fuel and has an energy peaking around 0.6 MeV. The neutron flux resulting from the alpha activity in the glass is taken into account when designing the canister shielding.

Modified spent fuel flasks, e.g. CASTOR 2A, TN 1300 and TN 12.2, among others, are well-suited for transportation of vitrified HLW (18, 19).

**Radioactive waste**

Any material containing or contaminated with radionuclides at concentrations greater than the values that the competent authorities would consider acceptable in materials suitable for uncontrolled use or release, and for which there is no foreseen use.

**Radioactivity**

Process whereby certain nuclides undergo spontaneous disintegration in which energy is liberated, generally resulting in the formation of new nuclides. The process is accompanied by the emission of one or more types of radiation, such as alpha particles, beta particles and photons (electromagnetic radiation).

**Radiolysis**

The decomposition of material by ionising radiation.

**Reprocessing**

A chemical process, the purpose of which is to extract for further use uranium and plutonium from spent fuel; this operation also results in the separation of radioactive waste products.

**Separative work**

A term used in the process of isotope enrichment; it is in relation to the amount of work necessary to increase the isotopic content of natural uranium in uranium-235.

**Sievert (Sv)**

A dose equivalent unit used in radiological protection. Doses in Sv are obtained by multiplying doses expressed in Grays by appropriate factors that take into account the biological effects of the various types of radiation, the dose distribution within the body and any other necessary corrections. The SI systems and ICRP proposed the use of the sievert as a new dose equivalent unit replacing the rem (R);  $1 \text{ rem} = 0.01 \text{ Sv} = 10 \text{ mSv}$ .

**Specific activity**

The activity per unit mass or volume.

**Spent fuel**

Same as irradiated fuel.

**Storage**

The emplacement of waste materials with the intent and in such a manner that the material can be retrieved later.

**Thermal reactor**

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

hours later and rested on a 15 m deep sand bank. It was loaded with 30 packages of the 48 Y type filled with natural UF<sub>6</sub> (236 THM), and 22 packages type 30B which were empty.

The authorities decided to recover the packages. Due to bad weather the operation took more than one month (till 4 October 1984). The packages were slightly dented by the stormy sea but remained tight except for one package which was damaged at the valve. The protective covers surrounding the valves did not withstand the incident. The contents of the damaged package was checked and showed a loss of 50 kg U.

During the salvage operation, U and F analyses were carried out continuously but no abnormal concentrations were detected. As a result of this accident an important regulatory action was undertaken at the international level and particular attention was drawn to the chemical hazard of UF<sub>6</sub>. A new regulation for UF<sub>6</sub> transport is being reviewed by IAEA (9).

The main new provision prescribes an experimental test on the fire resistance of UF<sub>6</sub> packages (30 minutes at 800°C) and on its behaviour during an impact (crash from 1.8 m height). The latter provision will impose changes to the valve covers, which did not survive the accident. A parallel action has been undertaken by the European Community in order to regulate the stowing and the fastening of large packages inside the shiphold.

### ***Disposal***

The release or emplacement of waste materials without the intention of retrieval.

### ***Dose***

The mean energy imparted by radiation per unit mass of matter; it is expressed numerically in grays, symbol Gy, as the unit of radiation equal to the joule per kilogram.  $1 \text{ Gy} = 1 \text{ J/kg} = 100 \text{ rad}$ .

### ***Dose equivalent***

The amount of absorbed radiation per unit mass of matter; it is expressed numerically in sieverts, symbol Sv, as the unit of dose equivalent.  $1 \text{ Sv} = 100 \text{ rem}$ .

### ***Dose equivalent limits***

The radiation exposure limits recommended by the International Commission on Radiological Protection (ICRP).

### ***Fast reactor***

A reactor in which the majority of fissions are produced by neutrons having energies of the same order of magnitude as those which they possess when they are produced in the fission process; often quoted as FBR for fast breeder reactors.

### ***Fissile material***

Material capable of undergoing fission.

### ***Fission***

The splitting of a nucleus usually into two approximately equal fragments; the process is accompanied by the emission of neutrons and the release of energy. Neutron-induced fission is the most important, but fission of certain nuclides may occur spontaneously.

### ***Fission products***

Nuclides produced in fission, either directly or by the disintegration of the fission fragments.

### ***Fission reactor***

A reactor in which energy is generated by the fission of heavy nuclei; includes both thermal and fast reactors.

### ***Fuel cycle***

The various stages involved in supplying fuel for nuclear power reactors and any subsequent treatment and disposal operations; it includes uranium mining, milling and enriching, fabrication of fuel elements, their use in a reactor, fuel reprocessing, waste storage and waste disposal.

### ***Gamma-rays***

Electromagnetic radiation with short wavelength ( $10^{-8}$  to  $10^{-11}$  cm) which is emitted by the nucleus. The emission of gamma-rays accompanies the disintegration of many alpha or beta emitters. The penetrating power of gamma-rays is a function of their energy.

- (17) *Research and Development of Spent Fuel Casks and the Criteria for Seagoing Vessels carrying them*, International Conference on Nuclear Power and its Fuel Cycle, Salzburg, 1977.
- (18) IAEA (International Atomic Energy Agency) Technical Reports Series N°. 229 (1983), *Handling and Storage of Conditioned High Level Wastes*, Vienna.
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- (20) WARD, T., *et al.*, (1990), *The WIPP Transportation System, Dedicated to Safety, Waste Management '90*, Vol. 1, pp. 235-238.
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## Glossary of Terms

### ***Actinide series***

The group of elements from actinium (atomic number 89) to lawrencium (atomic number 103), which together occupy one position in the periodic table; the series includes the naturally occurring and man-made isotopes of thorium and uranium and of the higher actinides such as neptunium, plutonium and curium.

### ***Activation products***

Nuclides which have become radioactive in consequence of bombardment by neutrons or other nuclear particles.

### ***Alpha particle***

A positively charged particle emitted in the decay of some radioactive nuclei, especially nuclides of the actinide series; it consists of two protons and two neutrons and is identical with the nucleus of the helium-4 atom. It has a very low penetrating power and hence pure alpha emitters are almost exclusively only a hazard when taken into the body.

### ***Becquerel***

A unit of radioactivity, defined as 1 disintegration per second. Related to the old unit of Curie (Ci) by  $1\text{Ci} = 3.7 \times 10^{10}$  disintegrations/second;  $1\text{Bq} = 2.703 \times 10^{11}$  Ci.

### ***Beta particle***

An electron or positron emitted in the decay of some radioactive nuclei; It is only moderately penetrating.

### ***Biological shield***

Thick walls, usually of concrete, separating a source of radiation from personnel and designed to reduce the level of the radiation to an acceptable level; the term is applied particularly to the shield surrounding the core of a nuclear reactor with the object of absorbing neutrons and gamma radiation.

### ***Breeder reactor***

A reactor that creates more fissionable fuel than it consumes; The new fissionable material is created by capture in fertile materials of neutrons from fission.

## **7. Safety Records of Fuel Cycle Facilities**

The experience gained from plant operations constitutes a valuable basis for the analysis and further improvements of safety. The emphasis of this status report is directed to the nuclear and radiological safety of the different types of fuel cycle facilities. It is not within the scope of the report to present a complete compilation of the vast experience and all the safety records of the nuclear fuel cycle facilities. However, operating data should demonstrate what has been achieved in the protection of workers against radiation and in limiting and controlling radioactive effluent discharges from the facilities.

As in the 1981 status report, a review of relevant incidents which have occurred in fuel cycle plants is presented.

It has to be emphasized that the material of this section should be regarded as additional information to the reader of this report to provide some perspective. However the material presented certainly is not sufficient, without considering additional factors, to derive conclusions of the safety status of specific facilities or to make comparisons. Important additional factors are the type and throughput of radioactive material in the facilities, the specific design and operation of the plant and site characteristics.

**Fuel Fabrication (THM per year)**

Belgium	435	Japan	1,655 (LWR) + (MOX)
Canada	2,900	Spain	200
France	1,775	Sweden	400
Germany	1,540	United Kingdom	2,106
Italy	260	United States	3,625

**Spent Fuel (AFR/THM)**

France	11,380	Sweden	3,000
Germany	3,095	United Kingdom	11,700
Japan	97 (under construction) 3,000	United States	750 (15,000)

**Reprocessing (THM per year)**

France	2,258	United Kingdom	2,700 (Magnox, THORP, etc.)
Japan	1,010 (operation) 210 (under construction) 800		

### 7.1.3 Uranium dioxide fuel element fabrication

#### Springfields fuel manufacturing plant, the United Kingdom

Year	Average individual dose mSv
1986	3.30
1987	3.05
1988	2.45
1989	2.00

#### SIEMENS Uranium fuel element fabrication, Hanau, Germany

##### Radiation exposure in 1989:

	Average individual dose, mSv	Collective dose man mSv
External gamma dose	1.17	568
Effective dose from incorporation	1.07	618
Additional incorporation from abnormal operation		348

### 7.1.4 Mixed oxide fuel element fabrication

#### ALKEM-SIEMENS-MOX fuel fabrication, Hanau, Germany

Year	Average individual dose, mSv	Collective dose man Sv
1984	3.01	1.36
1985	2.15	1.16
1986	4.03	2.63
1987	3.81	2.66
1988	3.41	1.76

## Away from Reactor Storage of Spent Fuel

Country-Site	Storage capacity (THM)	Storage means
<b>France</b>		
Cadarache (Cascad)	180	Vault
La Hague	8,400	Pool
<b>Germany</b>		
Gorleben	1,500	Dry Cask
Karlsruhe	55	Pool
Ahaus	1,500	Dry Cask
Kreifswald	500	Dry Cask
<b>Japan</b>		
Tokai-Mura	97	Pool
<b>Sweden</b>		
(Clab)	3,000	Pool
<b>United Kingdom</b>		
Sellafield (Magnox)	3,500	Pool
Sellafield (LWR + THORP)	5,300	Pool
Sellafield (AGR)	2,200	Pool
Wylfa	700	Vault
<b>United States</b>		
Morris	750	Pool
(MRS)	15,000 (Planned)	Dry

## Reprocessing plants

Plant	Beginning of construction	Starting operation	Capacity THM/YR
La Hague (UP3)	1982	1990	800
La Hague (UP2 800)	1982	1993	800
Sellafield (THORP)	1985	1992	1200
Rokkasho Mura	1992 (planned)	1999 (planned)	800

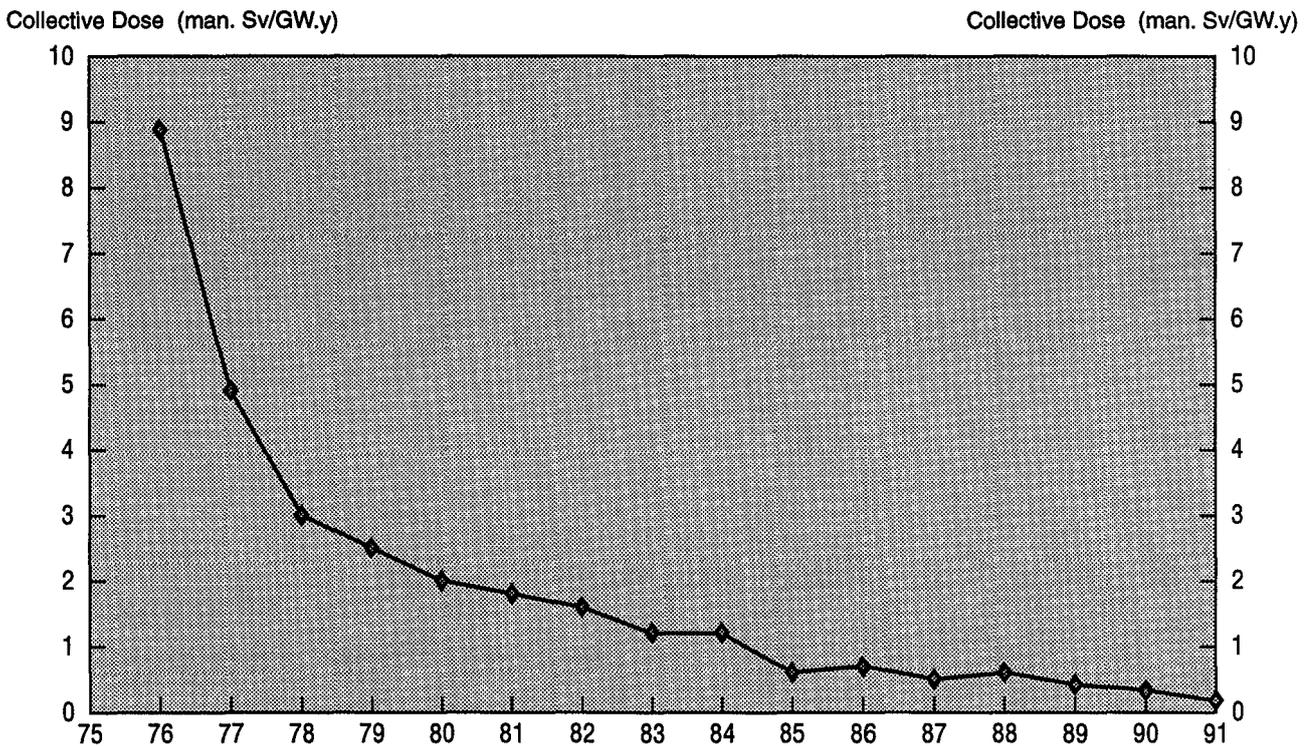
\* Construction of WAW plant was stopped in 1989.

### 7.1.5 Reprocessing

#### *La Hague:*

In 1991, the average individual dose at the reprocessing plants in La Hague was approximately 0.5 mSv per year, whole body dose; the total collective dose of La Hague was approximately 3.4 man Sv per year. These data are valid for all the personnel controlled by the radiation protection staff including personnel from subcontractors. The collective dose covers all activities on the site. A correlation of accumulated collective dose and plant throughput given by the energy equivalent gigawatt-year is shown in the following figure.

Collective dose of La Hague per gigawatt-year



#### *Sellafield:*

In the decade 1980-89, the external radiation dose to BNFL radiation workers at Sellafield has steadily decreased. In 1989 the average individual dose from external radiation was 3 mSv per year, the collective dose for all Sellafield's radiation workers summed up to 20 man Sv.

### Enrichment of Uranium (in SWU per year)

Country-Site	Production capacity (SWU per year)	Product process
<b>France</b>		
Tricastin	10,000,000	Diffusion
<b>Germany</b>		
Gronau	400,000	Centrifuge
<b>Japan</b>		
Ningyo-Toge	200,000	Centrifuge
Rokkasho-Mura	150,000	Centrifuge
	1,350,000 (Project)	Centrifuge
<b>the Netherlands</b>		
Almelo	1,200,000	Centrifuge
<b>United Kingdom</b>		
Capenhurst	950,000	Centrifuge
<b>United States</b>		
Paducah	11,300,000	Diffusion
Porthsmouth	8,000,000	Diffusion

### Fuel Fabrication

Country-Site	Production capacity (THM per year)	Product process
<b>Belgium</b>		
Dessel	35	MOX-LWR-FBR
Dessel	400	UO <sub>2</sub> - LWR
<b>Canada</b>		
Port Hope	900	Fuel Pellets & Bundles
Peterborough	1,000	Fuel Bundels
Toronto	1,050	Fuel Pellets
<b>France</b>		
Cadarache	25	MOX - FBR
Pierrelatte	500	UO <sub>2</sub> - LWR
Romans	750	UO <sub>2</sub> - PWR
<b>Germany</b>		
Hanau	40	MOX-FBR
Hanau	800	UO <sub>2</sub> - LWR
Karlstein	400	UO <sub>2</sub> - LWR
Lingen	300	UO <sub>2</sub> - LWR

**WAK, Karlsruhe reprocessing plant:**

Fiscal year	1985	1986	1987	1988	1989
<u>Collective doses (man Sv)</u>					
Total	1.11	1.36	1.72	1.30	0.96
Plant personnel	0.83	0.96	0.94	0.76	0.57
Personnel from contractors	0.28	0.40	0.78	0.54	0.39
<u>Average individual dose (mSv)</u>					
Total	2.0	2.53	2.90	2.04	1.53
Plant personnel	2.3	2.63	2.61	2.05	1.61
Personnel from contractors	1.5	2.33	3.18	2.03	1.42
<u>Maximum individual dose (mSv)</u>					
Plant personnel	19.2	27.0	19.6	11.2	8.0
Personnel from contractors	11.5	16.1	20.5	15.8	15.9

**Tokai reprocessing plant:**

Fiscal year	1985	1986	1987	1988	1989
<u>Collective doses (man Sv)</u>					
Total	0.84	0.73	1.55	4.46 <sup>1</sup>	1.59
PNC staff	0.24	0.28	0.33	0.29	0.24
Personnel from contractors	0.60	0.45	1.22	4.17 <sup>1</sup>	1.35
<u>Average individual dose (mSv)</u>					
Total	0.51	0.49	0.94	1.96	0.70
PNC staff	0.44	0.50	0.61	0.56	0.42
Personnel from contractors	0.54	0.49	1.10	2.36 <sup>1</sup>	0.79

1. Installation of the second pulse filter, replacement of the acid recovery evaporator.

## 7.2 Waste discharges

### 7.2.1 Uranium mining and milling

Annual uranium releases from Canadian uranium processing facilities in kilograms of uranium:

**CAMECO, Blind River UO<sub>3</sub> Refinery**

	1984	1985	1986	1987	1988	1989
Emissions to air	135	75.8	81.2	91.0	86.0	70.0
Emissions to water	33	12.9	6.6	9.0	13.0	11.0

## 9. Fuel Cycle Facility Listing of OECD Countries Status 1992

Sources: *Nuclear Engineering International, Special Edition (1990) World Nuclear Industry Handbook*  
*ESARDA - Analysis of Nuclear Fuel Cycle in E.C. Countries (1988)*  
*IAEA - Nuclear Fuel Cycle Information System (1987)*

### Uranium Ore Processing

Country-Site	Production capacity (THM per year)	Product process
<b>Australia</b>		
Olympic Dam	1,600	Ore - U <sub>3</sub> O <sub>8</sub>
Ranger	2,500	Ore - U <sub>3</sub> O <sub>8</sub>
<b>Belgium</b>		
Puurs-Englis	50	U Ex. H <sub>3</sub> PO <sub>4</sub>
<b>Canada</b>		
Stanleigh	600	Ore - U <sub>3</sub> O <sub>8</sub>
Key Lake	5,700	Ore - U <sub>3</sub> O <sub>8</sub>
Rabbit Lake	5,400	Ore - U <sub>3</sub> O <sub>8</sub>
Cigar Lake	5,400	In preparation
Midwest Joint Venture	1,600	In preparation
Minatco	2,000	In preparation
<b>France</b>		
Bessines	1,500	Ore - U <sub>3</sub> O <sub>8</sub>
Lodeve (St Martin du Bosc)	1,000	Ore - U <sub>3</sub> O <sub>8</sub>
Mailhac	500	Ore - U <sub>3</sub> O <sub>8</sub>
<b>Greece</b>		
Pananesti	150	Ore - U <sub>3</sub> O <sub>8</sub>
<b>Portugal</b>		
Urgeirica	170	Ore - U <sub>3</sub> O <sub>8</sub>

### 7.2.3 Uranium dioxide fuel element fabrication

Generally the gaseous effluents from uranium dioxide fuel element fabrication plants are very small (several grams of uranium per year, radioactivity in the order of 0.01 to 1 GBq alpha activity). The amount of uranium and radioactive daughter products discharged with liquid effluents is largely dependent on whether a wet or dry process is used for the conversion of UF<sub>6</sub> to UO<sub>2</sub>. The liquid discharges are very small for the dry conversion process and for wet conversion in the case of complete conditioning of liquid wastes to a solid CaF<sub>2</sub> waste product. Otherwise the amount of uranium in liquid discharges is dependent on plant throughputs and roughly of the order of some tenths of kilograms of uranium per year (some GBq per year).

### 7.2.4 MOX fuel element fabrication

Due to the high radiotoxicity of plutonium, the off-gas from MOX fuel element fabrication facilities is cleaned by several high-efficiency filters resulting in very small annual releases of radioactivity. As an example, for the new SIEMENS-MOX facility at Hanau, Germany the proposed limits for aerial discharges are 0.55 GBq per year activity and 3.7 TBq per year Rn-220 activity. Also for the MELOX-facility under construction at Marcoule, France, a similar low limit for the discharge of activity is foreseen.

Liquid waste is treated and consolidated to solid waste. For the discharge of remaining liquids or liquid from the uranium section of the MOX facility the proposed limits for discharge at the SIEMENS-MOX facility at Hanau are in the order of tenths of GBq per year.

### 7.2.5 Reprocessing

*La Hague:*

Atmospheric discharges, expressed in Bq (percentage of authorisation):

Year	Gas other than H <sub>2</sub>	H <sub>2</sub>	Halogenes	Aerosols
1982	5.1 × 10 <sup>16</sup> (10.6)	6.3 × 10 <sup>12</sup> (0.29)	1.5 × 10 <sup>10</sup> (14.0)	4.6 × 10 <sup>7</sup> (0.062)
1983	5.0 × 10 <sup>16</sup> (10.4)	8.3 × 10 <sup>12</sup> (0.38)	2.2 × 10 <sup>10</sup> (19.5)	3.9 × 10 <sup>7</sup> (0.053)
1984	2.7 × 10 <sup>16</sup> (5.6)	8.5 × 10 <sup>12</sup> (0.39)	2.8 × 10 <sup>10</sup> (25.2)	3.9 × 10 <sup>7</sup> (0.053)
1985	7.1 × 10 <sup>16</sup> (14.8)	3.3 × 10 <sup>13</sup> (1.50)	2.2 × 10 <sup>10</sup> (19.6)	6.9 × 10 <sup>8</sup> (0.932)
1986	2.9 × 10 <sup>16</sup> (6.0)	6.1 × 10 <sup>12</sup> (0.28)	1.10 × 10 <sup>10</sup> (10.4)	5.5 × 10 <sup>7</sup> (0.074)
1987	3.5 × 10 <sup>16</sup> (7.3)	1.5 × 10 <sup>13</sup> (0.68)	1.5 × 10 <sup>10</sup> (13.2)	5.5 × 10 <sup>7</sup> (0.074)
1988	2.7 × 10 <sup>16</sup> (5.6)	2.1 × 10 <sup>13</sup> (0.95)	2.2 × 10 <sup>10</sup> (19.6)	2.6 × 10 <sup>7</sup> (0.035)
1989	4.2 × 10 <sup>16</sup> (8.8)	2.5 × 10 <sup>13</sup> (1.14)	2.8 × 10 <sup>10</sup> (25.6)	2.3 × 10 <sup>7</sup> (0.031)
1990	6.3 × 10 <sup>16</sup> (13.1)	2.5 × 10 <sup>13</sup> (1.14)	1.9 × 10 <sup>10</sup> (17.4)	2.1 × 10 <sup>7</sup> (0.029)
1991	1.0 × 10 <sup>17</sup> (21.0)	2.8 × 10 <sup>13</sup> (1.27)	2.4 × 10 <sup>10</sup> (21.4)	2.6 × 10 <sup>7</sup> (0.035)

principle and the defence in depth concept. The full governmental surveillance on the licensing process of fuel cycle facilities – including siting, design, construction, commissioning, operation and eventual decommissioning – is structured differently in each country, although international and particularly IAEA activities can provide a significant degree of commonality. The follow-up of national and international regulations is achieved by inspectors who check the compliance with the site licence and monitor the licensee's safety performance.

Both internal and external safety aspects have to be taken into account in a comprehensive analysis of the plant safety.

The internal safety aspects to be discussed in a generic way are containment and ventilation, criticality and fire hazards; these are independent of the type of nuclear activity. The emphasis put on each of these aspects depends on the size of the facility and the quantity of nuclear materials involved. The external hazards must take into account the influence of major events (seismic events, floods, etc.) on the integrity of the plant and on the safety of the workers as well as that of the surrounding population.

Technical safety assessments are primarily concerned with the radiological safety. This includes the radioactive risk to man and his environment not only from direct radiation but also from environmental sources. The analysis of the mechanism for abnormal occurrences typical of nuclear facilities, such as criticality accidents, explosions, plant faults is also part of the technical safety assessment.

The man-machine interface is a new important aspect of safety management which ought to be taken into consideration. Though computer control has taken over many tasks in the plant formerly controlled by man, assessment of its reliability and fitness for purpose is essential.

Finally the assessment of accident consequences is also essential to any safety analysis since that aspect is directly connected to the public's perception of incidents and accidents. Recently the IAEA/NEA severity scale was developed for optional use by countries to inform the public about the seriousness of an incident by association with its severity levels which are analogous to the earthquake classification, i.e. the MSK scale.

Some countries have specific severity scales for fuel cycle facilities which have the advantage of being more specific, but which also might confuse the public who tend not to make a clear distinction between nuclear power plants and other nuclear facilities. A uniform severity scale with equivalent steps of impact from each type of nuclear facility is therefore commendable.

Special attention is being given in this report to the safety record of the nuclear fuel cycle facilities in the OECD countries. For reasons of historical overview all the significant incidents in the fuel cycle facilities have been reported according to their date of occurrence from 1950 up to 1990 including those reported in the 1981 report. The very good safety record of the civilian nuclear fuel cycle facilities in the OECD countries is a proof of the adequacy of the current regulatory and operating practices.

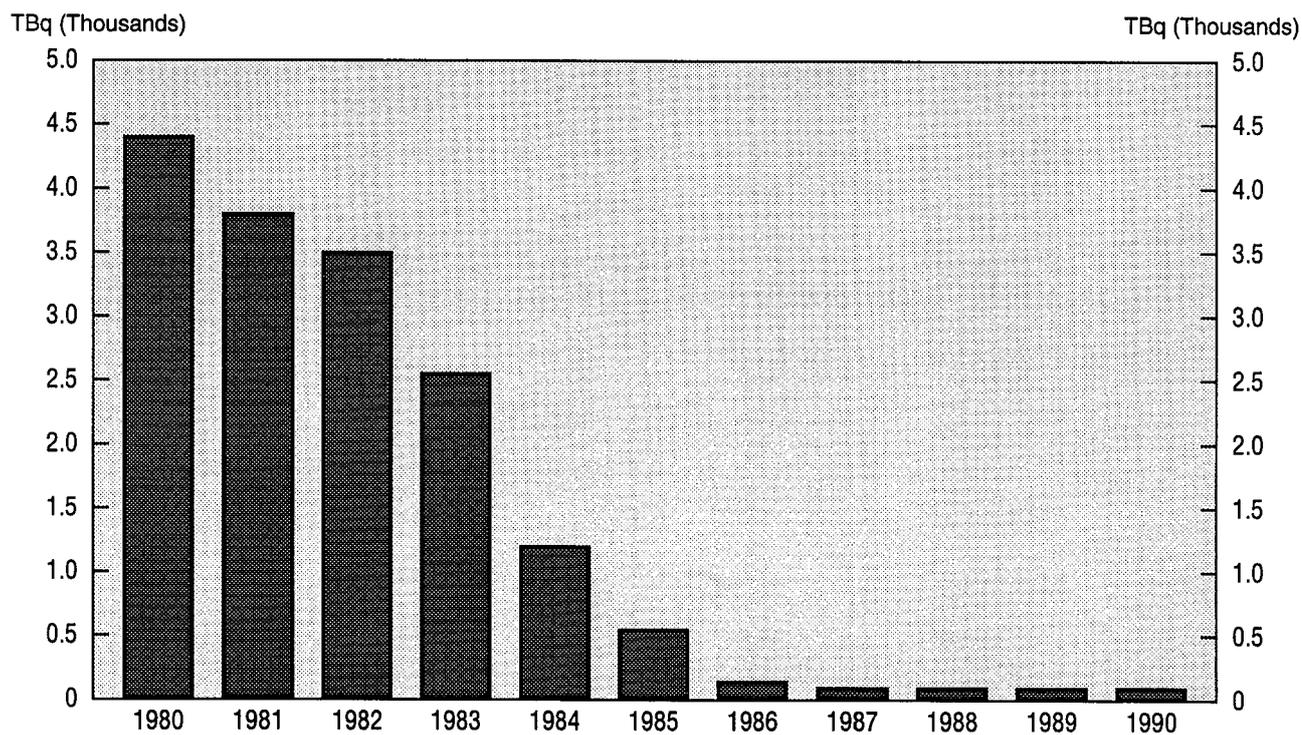
The steady decrease in personnel irradiation doses, and in effluent discharges into the surface waters and the air, is an illustration of the continued improvement in operating the facilities and the reduction of the radiological impact of the nuclear fuel cycle on the human environment.

### Liquid discharges, TBq

Year	H <sub>3</sub>	Total Beta other than H <sub>3</sub>	Strontium-90	Caesium-137	Total Alpha
1984	1,586	1,190	72	434	14
1985	1,062	587	52	325	6
1986	2,150	118	18	18	4.4
1987	1,375	89	15	12	2.2
1988	1,724	81	10	13	2.1

As the following figures demonstrate, a remarkable reduction of total alpha and beta radioactivity discharged by liquid effluents has been achieved in the last decade.

### Liquid discharges from Sellafield (total beta)



Storage of magnesium clad fuel resulting from the gas-cooled reactors is a declining activity which proved to be acceptable only for a limited period of time due to corrosion of the cladding. Reprocessing of Magnox fuel is therefore a necessity for which industrial plants exist in the United Kingdom and France.

The major hazard associated with Magnox reprocessing is fire; since the cladding is made of magnesium and the core of metallic uranium. During the mechanical decladding operations all possible precautions (active and passive) should be taken to avoid ignition and, in case a fire starts, to mitigate it. In 1987 an extensive regulatory safety audit took place at the Sellafield reprocessing plant and investigated all issues relevant to the safety of the plant and the environment.

In France, a silo filled with magnesium cladding hulls caught fire and therefore all storage facilities are now kept under a water cover. In the United Kingdom the Magnox swarf is being encapsulated in concrete for intermediate storage.

The liquid extraction carried out with TBP dissolved in a kerosene type solvent is the second potential source of fire for which adequate measures should be taken. However until now no large fires have been recorded in the Magnox reprocessing plants.

From a safety point of view, reprocessing of LWR fuel is the most complex part of the entire fuel cycle. Preventive measures against internal potential hazards, e.g. criticality, fire, explosion, corrosion, loss of containment, loss of cooling and mechanical damage, have to be taken into account during the design, construction and operation phases. Recently attention has turned to the external hazards, particularly seismic events and aircraft crashes. However, appropriate site selection and other measures could prevent such hazards to a satisfactory level.

The main potential hazard associated with industrial reprocessing lies in the fact that the nuclear materials are processed in the dispersible form of solution and powder and that industrial quantities of flammable solvent are used. However, only a small fraction of radioactive species is contained in the off-gas and the ventilation, and filtration systems are designed to cope with this. Chemical decomposition of organic reagents, e.g. TBP, degraded solvents, hydrazine, formaldehyde and production of radiolytic gases ( $H_2$ ), are the main explosion risks occurring in the PUREX process. The risks are carefully analysed in order to ensure process control systems capable of detecting and arresting the progress of abnormal events, and further to limit the consequences of an accident to the plant itself. The basic design of the plant must incorporate all measures to avoid criticality in every compartment, for example by geometric constraint and by proper mass flow control.

Fire and explosion risks should be reduced as much as possible and below an allowable value. Its spreading should be restricted by passive measures, e.g. compartmentalisation, use of incombustible structural materials, and mitigated by active measures, e.g. inertisation of equipment, fire-extinguishing system.

Loss of containment is a safety concern which is countered by choosing corrosion resistant materials and by surrounding the nuclear materials by a multi-barrier system consisting of the equipment itself, the containment building and a dynamic barrier with gradual ventilation pressure depression.

Corrosion resistance of the materials is a very important item in the assessment of the lifetime of the equipment, especially when very corrosive reagents, such as concentrated acids and redox

### Liquid discharges:

The WAK is connected to the Karlsruhe Research Center in respect of the treatment of liquid effluents and the conditioning of waste; therefore, no specific data on liquid discharges are available for the WAK alone.

### Tokai reprocessing plant:

#### Radioactivity released to the air

Nuclide	Annual	Activity, Bq				
		1985	1986	1987	1988	1989
Krypton-85	8.9E + 16	1.0E + 16	1.3E + 16	1.2E + 16	2.7E + 15	9.8E + 15
H <sub>3</sub>	5.6E + 14	2.8E + 12	2.7E + 12	3.7E + 12	2.5E + 12	3.7E + 12
Iodine-129	1.8E + 09	1.0E + 09	2.3E + 08	1.4E + 08	8.5E + 07	2.4E + 08

#### Radioactivity released to the sea

H <sub>3</sub>	1.9E + 15	2.6E + 14	2.4E + 14	2.6E + 14	7.4E + 13	2.4E + 14
Iodine-129	2.7E + 10	8.9E + 07	6.7E + 06	2.0E + 06	nd	1.3E + 07
Total $\alpha$	4.1E + 08	1.1E + 07	1.9E + 06	1.8E + 07	4.1E + 05	1.1E + 07
Total $\beta$	9.6E + 11	3.6E + 07	4.1E + 07	3.7E + 07	8.5E + 06	7.6E + 06
Strontium-90	3.2E + 10	2.3E + 06	2.5E + 07	8.5E + 06	nd	nd
Caesium-137	5.5E + 10	7.8E + 07	1.7E + 08	1.5E + 08	8.5E + 07	4.0E + 07
Plutonium( $\alpha$ )	2.3E + 09	4.8E + 06	9.3E + 06	1.2E + 07	4.1E + 06	4.8E + 06

nd = not detectable  
Total  $\alpha$  and  $\beta$  excluding specified single nuclides

## 7.3 Major incidents

As in other areas, incidents and accidents have occurred in the nuclear fuel cycle despite all technical and operational precautions. In all OECD Member countries, these abnormal events are reported by the plant operator to the authorities. Care is taken to collect this information in order to improve safety in the facilities and to provide information to the public and scientific bodies. Since minor incidents are reported and collected in most countries, the public might conclude that the nuclear fuel cycle industry is plagued by a never-ending series of incidents. Certainly it has to be admitted that incidents have occurred and will occur in the future, but compared to other industrial sectors, the safety record of all stages of the commercial nuclear fuel cycle is impressively good.

To allow comparison of incidents and to improve information provided to the public, severity scales for the evaluation of incidents have been established in some countries. However, the following description of major incidents, which have happened in fuel cycle facilities, is not based on a systematic evaluation of the severity of their consequences.

## 8. Main Conclusions

Since 1981 at the appearance of the previous report on the Safety of the Nuclear Fuel Cycle, the fuel cycle industry has considerably expanded and some sectors have seen their activity multiplied by five. At present the growth rate in the OECD countries seems stabilized and will depend to a great extent on the general evolution of the nuclear power capacity throughout the world. The fuel cycle facilities in operation or under construction can draw upon data from a very extensive and well-documented safety record accumulated over the past 30 to 40 years in international and national publications by technical experts and safety authorities. The major choices with regard to processes have been based on economic and safety considerations.

In the front end of the fuel cycle, no important changes have occurred in the overall production methods leading from uranium ore extraction to the production of fuel elements. However, due to the economic competitiveness, more emphasis was put on the production of  $\text{UO}_2$  and MOX fuel for LWRs at the expense of metallic uranium for the gas-cooled reactors and of  $\text{UO}_2$ - $\text{PuO}_2$  fuel for fast reactors.

From a safety point of view, the attention of OECD Member countries has expanded to include the "in process" problems and worker dose preoccupation as well as the external impact of the facilities on the environment.

The production of mill tailings, which accompanies the extraction of uranium from ore, aroused much attention among the safety authorities and strict regulations have been enacted to reduce the radiological consequences to the environment and to stabilize geochemically the treated ore-residue bodies. The underground mining of rich uranium ores is still a challenge to the personal safety of the workers due to the high radon concentration in the air. The uranium conversion units, from  $\text{U}_3\text{O}_8$  to  $\text{UF}_6$  or vice versa, and the uranium enrichment facilities are well established large-scale industrial processes with very high throughputs. These operations do not constitute a significant radiological hazard, but involve an inherent toxic chemical hazard owing to the massive use of  $\text{UF}_6$  and fluorinated reagents. Such hazards are similar to those in the conventional chemical industry and their control is well-established and understood.

The impact of external hazards (such as earthquakes, fires) on the safety of the large enrichment plants, was recently examined carefully. This examination confirmed the adequacy of the initial safety design concepts.

With the continuing buildup of depleted  $\text{UF}_6$  stocks from some enrichment plants, increased attention is to be given to the safe storage of these residues or to their conversion to less hazardous chemical compounds, i.e.  $\text{UF}_4$ ,  $\text{U}_3\text{O}_8$ , etc. which can be stored without much risk for an extended period of time and which are not hazardous even in accidental conditions such as aircraft crashes, fires, etc.

alarm, presumably were exposed for about 15 seconds, which is roughly the interval between the first two pulses. It is apparent that exposures were limited by prompt evacuation.

The following corrective measures were adopted subsequently. Instead of relying upon valves for isolating equipment, transfer lines that may contain fissile material are actually disconnected. Only vessels that would be sub-critical when containing uranium-235 enriched uranium solutions are permitted.

*The Los Alamos Scientific Laboratory, 30 December 1958*

The next accident resulted from the concentration of plutonium in a solvent layer which was found in a large tank that was supposed to contain only lean aqueous-organic emulsion. A transient change of shape of the solvent layer when a stirrer was started established a criticality of short duration. The result was a fatality and two other significant exposures.

The accident occurred in an area where residual plutonium, usually about 0.1 gram per litre, and americium were recovered from dilute raffinate. Because the normal plutonium inventory was only 0.1 kg, solvent extraction was conducted in large closed tanks. As at Y-12, a material inventory was in progress and it was intended that the tanks be emptied and cleaned individually. Instead, residues and acidic wash solutions from four vessels were combined in a single 850-litre, 96.5 cm diameter tank; many interconnecting transfer lines made this possible. An excursion of  $1.5 \times 10^{17}$  fissions occurred when a stirrer in this tank was started.

As discussed later, a 20.3 cm thick, 160-litre organic layer floating on a dilute-aqueous solution contained 3.27 kg plutonium. It is presumed that the source of this plutonium was solids that had accumulated gradually in the tanks during the 7.5 years of operations and that the organic layer resulted from separation of the emulsion phases by added acids. The initial effect of the stirrer was to thicken the axial part of the organic layer sufficiently for super-criticality. Continued rapid stirring mixed the two phases, diluting the plutonium to a sub-critical concentration.

The operator, who was looking into the tank through a sight glass, received an exposure of 120 ( $\pm$  60) Gy and died 36 hours later. Two men who went to the aid of the victim received doses of 1.3 Gy and 0.35 Gy. There was neither damage to equipment nor contamination, although a shock displaced the tank support 10 mm. A radiation alarm 53 m away was activated and a flash of light was seen from an adjoining room.

The entire recovery plant, which had been scheduled for rebuilding after another six months of operation, was shut down immediately. After ultimate conversion to geometrically sub-critical equipment, the following corrective measures were adopted: written procedures and nuclear safety training were improved, unnecessary solution transfer lines were blocked, and auxiliary vessels such as vent tanks and vacuum buffer tanks were "poisoned" with borosilicate glass raschig rings. Periodic surveys with portable neutron detectors to locate abnormal plutonium deposits were instituted. The accident also led to more complete coverage of process areas by improved gamma-ray sensing radiation alarms.

- (18) Report on incidents in Sweden, presented at the annual meeting of CSNI Working Group on the Safety of the fuel Cycle, September 1.
- (19) Health and Safety Executive (1980), *The Leakage of Radioactive Liquor into the Ground at BNFL Windscale*, 15 March 1979.

*The Recuplex Plant, Hanford, 7 April 1962*

This incident occurred when liquid from a sump was collected in a 69-litre, 45.7 cm diameter vessel. The liquid, unidentified at the time, contained between 1,400 and 1,500 grams of plutonium in a volume of about 46 litres after the addition of lean solutions. The only significant exposures were 0.88 and 0.16 Gy received by personnel at distances of about 2.1, 3.2 and 7 m respectively from the excursion.

The site was a plutonium recovery plant in room-sized glove boxes to prevent external contamination. The vessel in which the excursion occurred was normally used for transfer of a dilute side stream for solvent extraction columns to a secondary recovery process, similar to the raffinate-treatment process of the Los Alamos accident. Apparently the concentrated solution had overflowed from a geometrically sub-critical tank and was sucked into the 45.7 cm diameter vessel through a temporary line used for clean-up operations that were still in progress. A total yield of  $8.2 \times 10^{17}$  fissions occurred over 37 hours, with about 20 per cent of the energy released in the first half hour. An initial pulse of approximately  $10^{16}$  fissions was followed by smaller pulses for about 20 minutes, after which boiling occurred, ultimately distilling off enough water to stop the reaction.

The initial pulse, accompanied by the usual blue flash, triggered a radiation alarm, and the area was evacuated promptly, presumably before a second pulse. A unique feature of the analysis of the events was the use of a small, remote-controlled robot developed for handling irradiated fuel. By means of this device, the excursion site was located, meters were positioned and read, and valves were operated.

A new plant to replace Recuplex had been authorised before the accident, and operations were not resumed until it became available. In the modern plant, vessels that are not sub-critical by geometry usually contain neutron absorbers. The system is adaptable to a variety of uses without improvisation, and equipment is easier to keep clean. It is recognised that the flexibility needed in this salvage plant requires special effort to maintain realistic, up-to-date written procedures.

*Wood River Junction Plant, 24 July 1964*

This accident was initiated when concentrated enriched uranium solution was inadvertently poured into a 45.7 cm diameter (75 litres) tank. The first of two excursions resulted in a lethal exposure and the second, about 2 hours later, was primarily responsible for two other significant radiation doses.

Start-up difficulties in this plant for recovering highly enriched uranium from scrap led to an unusual accumulation of trichloroethane (TCE) solution of low uranium concentration. Small amounts of uranium were recovered by tedious hand agitation of the TCE with sodium-carbonate solution. An easier process was improvised, in which the TCE was treated in the 45.7 cm diameter tank that had been intended only for the make-up of sodium-carbonate solution used in the normal recovery process. Neither the plant superintendent nor one of three shift supervisors was aware of this practice. Meanwhile, solutions of unusually high uranium-235 concentration, resulting from clean-out of plugged equipment had been stored in 11-litre, 12.7 cm diameter bottles identical to those that contained the contaminated TCE. Apparently, a bottle of the concentrated solution was mistaken for TCE and was poured into the sodium-carbonate solution being stirred in the make-up tank. The sock from a single pulse of  $\sim 10^{17}$  fissions knocked the operator onto the floor and splashed part of the solution out of the tank. A flash of light was observed. The victim received an exposure estimated to the 100 Gy and died 49 hours later.

### *Radiological consequences*

The accident and the subsequent evacuation from the controlled zones caused no physical damage to the operating and maintenance personnel.

The systematic air sampling carried out every few hours showed no general contamination of the buildings except for slight air activity in one room of one building. This disappeared as soon as the ventilation system was restarted. The associated surface contamination was cleaned up during the day of 17 April. The air samples also demonstrated that there had been no discharge of radioactivity from the stacks.

### *Restarting conditions*

On 21 April 1980, power was supplied to the plant in the following manner:

- the power supply for the normal operation of the factory was provided from the EDF 90 kV grid by a temporary connection;
- a first emergency power supply was provided by a set of fixed generators capable of maintaining the power necessary for the safety of the installation;
- a second emergency power supply was provided by a set of mobile generators and connection through a temporary line to the EDF 15 kV country grid.

In these conditions, and after regulatory inspection of the system, restart of the plant was authorised on 22 April 1980.

neutron monitors should be installed on all vessels which are not safe by shape and that provision be made to enable listed plant items to be emptied completely.

*The Idaho Chemical Processing Plant (3, 4), October 17, 1978*

This incident occurred in the first cycle solvent extraction system which was recovering highly enriched uranium from salvage solutions. A gradual decline in the concentration of aluminium nitrate solution in the aqueous scrub stream feeding column permitted a build-up of uranium concentration in the base of the column, to 21 or 22 grams uranium per litre. The mass of uranium in the scrubbing column increased markedly and criticality occurred in the base of the column producing an estimated  $3 \times 10^{18}$  total fissions. The criticality occurred in a heavily shielded and suitably ventilated cell. There were no injuries; no release from a breach of containment; and, no damage to equipment.

The probable cause of this incident was the failure of the management and review systems to provide for or to prevent the deterioration of administrative controls and alarm instrumentation, both of which had previously been identified as being required.

### **7.3.2 Release of $UF_6$**

*$UF_6$  accidental release in France (6), 1 July 1977*

*i) The incident*

The incident happened at Pierrelate, France, involving a cylinder containing 8,827 kg of liquid  $UF_6$  at around  $95^\circ C$  after a liquid phase sampling. The cylinder was resting on a berth, close to the ground next to the sampling building, its valve being at the lowest position. During handling, a mishap occurred and the valve, still connected to the sampling building via a flexible tubing, broke flush with the cylinder wall leaving an opening slightly under one inch diameter: its area was  $3.84 \text{ cm}^2$ .

The release of toxic materials took place over almost one hour. Using fire-fighting equipment,  $13.5 \text{ m}^3$  of water was sprinkled on and around the source over about 1/2 hour. Additionally, 600 kg of liquid carbon dioxide were sprinkled 15 to 20 minutes after the initial burst.

A total of 7,106 kg of  $UF_6$  was released to the atmosphere according to a weighing of the cylinder plugged at the end of the release which showed a residual amount of 1,721 kg  $UF_6$ .

*ii) Data collected*

The following three points are worth stressing (note that wind velocity was 10 m per second):

- the amount of uranium detected in the atmosphere was only a small fraction of the  $UF_6$  vapourised;
- hydrogen fluoride from  $UF_6$  hydrolysis could be traced as expected in an area extending 15 km downwind;
- the fluorine to uranium weight ratio in the ground combination became significantly large beyond 1,000 m.

At the locations where the highest concentrations were detected, chemical dangers were small. As for the radioactive hazard associated with uranium deposited on the ground, a dose of 1,50 mSv

instructions to float off solvent and retain a heel were therefore not carried out before the transfer to the sea tanks commenced at about 21:15 hours on November 10th.

Transfer proceeded normally until about 00:45 hours on November 11 when gamma radiation monitors on the transfer pipework alarmed and the transfer was terminated. The high-radiation levels on pipework were reduced by washing through into the sea tank.

After this transfer, sea tank No. 1 was found to contain about 166,5 TBq of beta activity mostly contained in a thin layer of solvent and crud floating on top of a much less active aqueous liquor. Despite the temporary high-radiation levels on pipework, no person received a significant radiation dose. There remained, however, an enhanced dose-rate of about 0.05 mSv per hour around the sides of sea tank No. 1 which caused the 7.5  $\mu$ Sv per hour limit for the nearby boundary of the Controlled Area to be exceeded and required the boundary to be temporarily extended.

It was decided to discharge most of the aqueous liquor in sea tank No. 1 to sea before transferring the much smaller quantity of aqueous liquor, solvent and crud remaining up a 5/7.5 cm emergency return line to a tanks in B211. This was duly carried out and resulted in about 0.2 TBq of beta activity being sent to sea.

The transfer of the remaining liquor was then carried out which involved pumping it through the normal 25.4 cm pipework to the sea discharge lines for some distance and then up the 5/7.5 cm return line to B211. After transfer the pipework was washed through into B211 but radiation levels on the 25.4 cm pipework remained high at about 20 mSv per hour. It was considered that this was due to the low flow rate achievable in pumping up the 5/7.5 cm line (about 4.5 m<sup>3</sup> per hour) being inadequate to flush remaining crud and solvent from the 25.4 cm pipework.

It was decided to flush this material to sea by discharging a sea tank (at the normal flow-rate of about 400 m<sup>3</sup> per hour) through the 25.4 cm pipework and down No. 1 old sea line. This was carried out in the early hours of Sunday, November 13 and thereby reduced radiation levels on the pipework gamma radiation monitor to about 6 mSv/h. During the flushing operation the radiation level at the sea discharge valve exceeded 0.05 Sv/h, but prompt evacuation prevented workers stationed there from receiving any significant radiation doses.

For reasons not yet completely understood the radiation level on the pipework rose to high level (of some mSv/h) on a number of further occasions and further flushing was carried out that day to reduce them with the last being at about 8 p.m.

A further flushing of the pipework was carried out late on Wednesday, November 16th, after the gamma radiation level on the pipework rose to several mSv/hour. This process reduced the level to about 4 mSv/hour.

A solvent slick was observed on the surface of the sea on Friday, November 18th, but before it could be dispersed with detergent the wind changed to on-shore and, on Saturday, November 19th, deposited it over a small length of the shoreline adjacent to the site.

A total of  $5.9 \times 10^{13}$  (1,600 Ci) including  $4.5 \times 10^{13}$  (1,214 Ci) ruthenium-106 had been released to the sea. Maximum irradiation levels were 270 mSv per hour on dried organic material deposited on the beach. A great number of measurements resulted in dose rate  $> 0.1$  mGy/h. The access to the beaches was temporarily closed.

When the operator noted that he was unable to add more UF<sub>6</sub> to the cylinder, he investigated and discovered that the wheel was off the platform. After the cart and cylinder were repositioned, the scale was unable to record the actual weight of the cylinder because the weight exceeded the dial indicator range. The cylinder at this point weighed more than 13.4 tonnes, the limit of the dial indicator. Later estimates placed the weight in excess of 14 tonnes; the fill specification for this cylinder is 12.5 tonnes. The operator adjusted the tare-compensating mechanism on the scale to permit observation of weight loss while he attempted to evacuate the excess material by vacuum back into the cold traps. This evacuation attempt is in accordance with procedures and accepted practice.

On the subsequent shift the next operator concluded that the material was no longer being evacuated, presumably because the contents of the cylinder had cooled and solidified. The operator and the assistant shift-supervisor moved the cylinder to a steam chest for the purpose of heating the cylinder to liquefy the contents and facilitate later evacuation. The weight of the contents at this point is not known, but the cylinder definitely contained more than the fill specification of 12.5 tonnes. In any case, the cylinder was definitely overfilled. Sequoyah Fuels Corporation operating procedures prohibit the heating of an overfilled cylinder. Placing the overfilled cylinder in the steam chest for heating was clearly a violation of these procedures.

Approximately two hours after heating began, the cylinder ruptured in the steam chest. A 1.2 meter lengthwise rupture occurred along the top of the cylinder.

The force of the explosion damaged the steam chest enclosure. The escaping UF<sub>6</sub> rapidly reacted with moisture in the air to form uranyl fluoride and hydrofluoric acid. The resulting vapour cloud of these materials was carried south by southeast by a wind gusting to 40 km/h. The cloud enveloped the process building, and the acidic vapour caused the death of an operator who was working approximately 21 meters from the cylinder. The vapour was drawn into the plant ventilation system. Approximately 40 workers in the building evacuated to an upwind location on site, some passing through the cloud.

The injured workers were transported by fellow workers to nearby hospitals for treatment. The general population downwind was contacted and advised to evacuate and report to hospitals for examination.

The following factors were identified as the primary contributors to the accident:

- The cylinder was overfilled because it was not placed fully on the scales. The fill bay and associated equipment were not designed to prevent improper positioning of cylinders in the bay so that the cylinder would not be on the scales. The fill bay was not designed to accommodate 14-ton cylinders.
- The time required for filling the cylinder was long enough to allow partial solidification of the UF<sub>6</sub> which inhibited product removal from the cylinder.
- The precise weight of the cylinder was not readily determinable after it was overfilled.
- There was no secondary or alternative way to measure the quantity of material in a cylinder being filled.
- Employees violated company procedures when they heated an overfilled cylinder. Workers, including line-management personnel, had not been trained in regard to company procedures. Procedural controls such as checklists or approval points were not an integral part of plant operations.

This incident, caused by a defective valve, resulted in a high contamination of the soil. The maximum dose rate level was 0.01 Gy per hour at one meter from the soil.

Preventive measures taken after the incident included checking of all stopcocks and the installation of control systems automatically closing the draincocks when the filter effluents have reached the upper level in the tank.

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On November 26, 1977 a contamination incident occurred in a plutonium conditioning and storage plant at La Hague. Bellows of handling tongs became disconnected from the opening ring of one of the two twin boxes used for plutonium dioxide conditioning. This incident resulted in a rapid air intake in the glove boxes, so that for a few seconds there was no negative pressure in the glove boxes. The pressure regulation system was too slow to prevent this increase of pressure. An overpressure was even temporarily found in one of the two boxes. An unsuspected leakage of one box thus caused an air contamination of 2,000 MPC-h (Maximum Permissible Concentration) in the room. Nobody was involved in this incident.

Following this incident a research and development programme was set up to:

- improve the quality of the elastomer constituting the bellows of the tongs and the mechanical and thermal protection of these bellows;
- decrease the inertia of the pressure regulation system to improve the response to any sudden pressure variation.

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At the La Hague fuel reprocessing plant, two other incidents have been reported, both occurring at the High Activity Oxide (HAO) building.

The first incident happened on 3 February 1978 in the area of a sampling circuit associated with a high radioactive process tank. This circuit comprises an air lift system which lifts the solution from the tank to a pot where samples are taken and sent to the laboratory. The lower end of the air lift tube should always be kept immersed in a dead volume of liquid at the bottom of the tank. The air lift system was accidentally left in service; the air flow generated by the air lift system carried radioactive aerosols in the upper circuit through the reflux column and the filter.

The consequences were high contamination of the off-gas circuit, normally protected by the reflux column and the filter, resulting in high contact radiation but, fortunately, in low dose exposure of the personnel. The causes of this incident could probably be attributed to design and/or construction defects which allowed the liquid to drop below the level of the air lift pipe and to a lack of co-ordination between the laboratory staff and the plant operators during sampling. Preventive measures taken after this incident included a review of the operating instructions concerning the air lift systems.

The second incident happened on May 10, 1978. This was a leakage from lead-shielded valves on the liquid effluent line from the spent fuel chemical decladding plant. For unknown reasons the alarm connected to the effluent gauge did not work. About 2 m<sup>3</sup> of active liquid leaked through the shielding and spread over the cell floor. Another radiation detector gave the alarm late.

for approximately six weeks. Operation was resumed without the use of the anion exchange contactor and will continue indefinitely in this manner.

ii) Analysis of evidence and findings

*Examination of the anion exchange contactor*

Examination of the contactor revealed that the stainless steel section in the lower portion of the contactor showed a uniform pipe diameter expansion caused by internal pressures. The expansion involved an increase in diameter from a nominal 11.4 cm outside diameter to 12.4 cm outside diameter, with a decrease in wall thickness. The schedule-40 pipe components showed no significant enlargement.

One flange gasket in the main column above the feed tee was missing completely, and a pair of Teflon gaskets on the XAW tee connection was ruptured. Screens and screen supports in two lines in the lower section of the contactor were ruptured or distorted, and three screens from widely different portions of the contactor showed discoloration, indicating exposure to high temperature. There was no evidence of any pipe rupture or pressurisation of any other section of the ion exchange loop. Instrument records support the contention that the elapsed time from the beginning of the pressure increase to the final pressure release was less than one minute.

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An inventory of the resin after the incident established that approximately one half of the resin remained in the column. A considerable quantity of resin was recovered from the floor of the main process area. The distribution of the resin was in a generally circular pattern of 2.4 to 3 meters radius, centering on the ion-exchange contactor.

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A liquid sample withdrawn from the feed tank for the anion exchange contactor showed the presence of 4.9 grams per litre of  $\text{Cr}^{3+}$  and 1.7 grams per litre of  $\text{Cr}^{6+}$ . The solution also contained 16 grams per litre of plutonium, of which 15.2 grams per litre was in the +6 valence state.

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Pressurisation of the anion exchange contactor is believed to have been caused by chemical degradation and oxidation of the plutonium-loaded anion exchange resin. It is suggested that the degradation of the resin may have been triggered by the inadvertent addition of sodium dichromate to the system, and it is estimated that 1 to 2 hours of exposure to dichromate at 85°C would have been sufficient to produce the kind of resin degradation observed in samples after the incident.

Evidently the prolonged period of high plutonium loss to the XQW waste was directly related to the presence of dichromate to the system.

- the heating rate has been lowered so that any organic reacting in the denitrators would, upon decomposition, evolve gas at a relatively low rate;
- the air flow through the denitrator off-gas has been increased to ensure that any evolved organic vapour has a concentration below its explosive limit;
- new process instrumentation has been installed to determine whether or not the new limitations on denitration conditions are being met. The new instrumentation also provides for a more rapid response to violations of critical controls, either manually (following an alarm) or automatically.

*Explosion of an off-gas scrubber in the scrap recovery at the Siemens uranium fuel element fabrication, Hanau, Germany (16), 12 December 1990*

On December 12, 1990, at the fuel fabrication plant for uranium fuel elements in Hanau, Germany an explosion occurred in a scrubber. The spray scrubber for off-gas cleaning was part of the installations for uranium recycling and treatment of liquid waste streams.

The lower part of the scrubber column built of steel was ripped and demolished, the bottom was thrown down and the upper part built of PVC was broken into many pieces; also bolts and flanges were torn. The housing of the loop pump was smashed into small missiles. The adjacent storage tanks, pipings and switch cabinets were deformed by the pressure wave. Some storage tanks were penetrated by missiles and also the roof of the hall got some small holes.

From the destroyed or damaged storage tanks about 800 kg low enriched uranium leaked to the floor mainly as a solution. The liquids containing uranium were kept within the provided sections at the floor of the hall. The off-gas filtering system was not affected. No radiation or emission of radioactivity to the environment occurred. Two workers were injured, one of them severely.

*Investigation of the incident causes*

The scrubber had to treat the off-gas of several systems:

- dissolvers, for recycling of uranium oxide, delivering mainly nitrous gases;
- furnaces, for calcination of ADU filter cake and for drying clarification mud, delivering ammonia and also organic carbon and fluoride;
- some storage and precipitation tanks, and other components also delivering ammonia.

So in normal operating conditions in the scrubber pool there was present an ammonium nitrate solution with parts of ammonium nitrite, fluoride and organic compounds.

Due to a failure of the liquid level control, the ammonium nitrate concentration in the scrubber liquid increased because of evaporation of the water into the off-gas. Probably the formation of slurry, or crystallisation, occurred in consequence. Obviously the explosion was initiated by the hot-running loop pump. The rapid thermal decomposition of ammonium nitrate destroyed the pump and struck back to the sump of the scrubber where an explosion also took place and the remaining scrubber pool liquid was thrown upwards.

The typical residue samples taken from several places in the scrubber wreck and in the loop contained ammonium nitrate up to 50 per cent (weight) and about 10 per cent fluoride as well as about 1 per cent nitrite. The values in the solution and in the slurry were certainly lower. The nitrite might have stimulated the starting reaction because of the lower decomposition temperature. The fluoride and the uranium content as well as the organic compounds might have had a catalytic effect.

protection of personnel and the consequences of the incident. The company also conducted an inquiry and provided all the technical data to the NII. On the balance of evidence, some tentative conclusions may be reached on the mechanisms which could have led to the escape of ruthenium-106 into the B204 working area.

- Exothermic reactions took place in a constant volume feeder CVF 111 between high-temperature insoluble fission products containing large quantities of ruthenium-106, zirconium fines and other solids and the acidified butex ( $\beta$ -dibutoxy-diethyl-ether) leading to the decomposition of the butex and, possibly, ignition of the zirconium.
- Air samples taken from the plant stack indicated that a release of ruthenium-106 to atmosphere of about 370 GBq (10 Ci) had occurred. This was equivalent to less than one fifth of the then appropriate daily derived working limit (DWL) for the stack. No airborne activity was detected above background levels downwind at the site boundary or beyond.

Air samples taken in the working area at the time of the incident indicated significant levels of airborne activity; the highest levels were approximately 100 times the maximum permissible concentration in air for active areas recommended by ICRP.

The seventh floor was found to have the highest levels of surface contamination. In the plant operating area adjacent to a constant volume feeder vessel and can trough, the surface contamination levels were greatly in excess of the DWL for surface contamination in active areas.

All 35 persons in the head end building at the time of the incident were contaminated on the body. The external contamination produced no adverse health effects. The amounts inhaled have been measured on a whole body monitor for all 35 men. One man was estimated to have up to about 1.5 MBq in his lungs, the others to have amounts ranging from 0.35 kBq to 185 kBq. The man with the highest incorporation had an estimated dose commitment in 50 years of approximately 10 Sv. The men in the next most highly irradiated group (four men) had estimated dose commitments ranging from 0.15 to 0.3 Sv and the remaining men had estimated dose commitments of up to 0.15 Sv. No adverse medical effects have been observed to date as a result of the internal exposures.

Proposals were made to reduce the likelihood of a recurrence of butex/solids reactions entailing monitoring the arising of solids in irradiated oxide fuel processing and removal of such solids in plant process vessels. Recommendations were made for improving the protection of operators in B204 and especially on:

- the sealing arrangements for penetration between the working area and the cell;
- the ventilation arrangements which led to the escape of radioactive materials from the processing cells;
- the beta-in-air monitors which did not appear to be adequate in type and number to give unambiguous warning of high airborne contamination levels;
- the emergency arrangements which did not deal adequately with the situation. The lack of suitable emergency arrangements was evidently due to the fact that there had been no similar incident on the site, or, as far as is known, anywhere else.

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It was decided to discharge most of the aqueous liquor in sea tank No. 1 to sea before transferring the much smaller quantity of aqueous liquor, solvent and crud remaining up a 5/7.5 cm emergency return line to a tanks in B211. This was duly carried out and resulted in about 0.2 TBq of beta activity being sent to sea.

The transfer of the remaining liquor was then carried out which involved pumping it through the normal 25.4 cm pipework to the sea discharge lines for some distance and then up the 5/7.5 cm return line to B211. After transfer the pipework was washed through into B211 but radiation levels on the 25.4 cm pipework remained high at about 20 mSv per hour. It was considered that this was due to the low flow rate achievable in pumping up the 5/7.5 cm line (about 4.5 m<sup>3</sup> per hour) being inadequate to flush remaining crud and solvent from the 25.4 cm pipework.

It was decided to flush this material to sea by discharging a sea tank (at the normal flow-rate of about 400 m<sup>3</sup> per hour) through the 25.4 cm pipework and down No. 1 old sea line. This was carried out in the early hours of Sunday, November 13 and thereby reduced radiation levels on the pipework gamma radiation monitor to about 6 mSv/h. During the flushing operation the radiation level at the sea discharge valve exceeded 0.05 Sv/h, but prompt evacuation prevented workers stationed there from receiving any significant radiation doses.

For reasons not yet completely understood the radiation level on the pipework rose to high level (of some mSv/h) on a number of further occasions and further flushing was carried out that day to reduce them with the last being at about 8 p.m.

A further flushing of the pipework was carried out late on Wednesday, November 16th, after the gamma radiation level on the pipework rose to several mSv/hour. This process reduced the level to about 4 mSv/hour.

A solvent slick was observed on the surface of the sea on Friday, November 18th, but before it could be dispersed with detergent the wind changed to on-shore and, on Saturday, November 19th, deposited it over a small length of the shoreline adjacent to the site.

A total of  $5.9 \times 10^{13}$  (1,600 Ci) including  $4.5 \times 10^{13}$  (1,214 Ci) ruthenium-106 had been released to the sea. Maximum irradiation levels were 270 mSv per hour on dried organic material deposited on the beach. A great number of measurements resulted in dose rate  $> 0.1$  mGy/h. The access to the beaches was temporarily closed.

neutron monitors should be installed on all vessels which are not safe by shape and that provision be made to enable listed plant items to be emptied completely.

*The Idaho Chemical Processing Plant (3, 4), October 17, 1978*

This incident occurred in the first cycle solvent extraction system which was recovering highly enriched uranium from salvage solutions. A gradual decline in the concentration of aluminium nitrate solution in the aqueous scrub stream feeding column permitted a build-up of uranium concentration in the base of the column, to 21 or 22 grams uranium per litre. The mass of uranium in the scrubbing column increased markedly and criticality occurred in the base of the column producing an estimated  $3 \times 10^{18}$  total fissions. The criticality occurred in a heavily shielded and suitably ventilated cell. There were no injuries; no release from a breach of containment; and, no damage to equipment.

The probable cause of this incident was the failure of the management and review systems to provide for or to prevent the deterioration of administrative controls and alarm instrumentation, both of which had previously been identified as being required.

### **7.3.2 Release of UF<sub>6</sub>**

*UF<sub>6</sub> accidental release in France (6), 1 July 1977*

*i) The incident*

The incident happened at Pierrelate, France, involving a cylinder containing 8,827 kg of liquid UF<sub>6</sub> at around 95°C after a liquid phase sampling. The cylinder was resting on a berth, close to the ground next to the sampling building, its valve being at the lowest position. During handling, a mishap occurred and the valve, still connected to the sampling building via a flexible tubing, broke flush with the cylinder wall leaving an opening slightly under one inch diameter: its area was 3.84 cm<sup>2</sup>.

The release of toxic materials took place over almost one hour. Using fire-fighting equipment, 13.5 m<sup>3</sup> of water was sprinkled on and around the source over about 1/2 hour. Additionally, 600 kg of liquid carbon dioxide were sprinkled 15 to 20 minutes after the initial burst.

A total of 7,106 kg of UF<sub>6</sub> was released to the atmosphere according to a weighing of the cylinder plugged at the end of the release which showed a residual amount of 1,721 kg UF<sub>6</sub>.

*ii) Data collected*

The following three points are worth stressing (note that wind velocity was 10 m per second):

- the amount of uranium detected in the atmosphere was only a small fraction of the UF<sub>6</sub> vapourised;
- hydrogen fluoride from UF<sub>6</sub> hydrolysis could be traced as expected in an area extending 15 km downwind;
- the fluorine to uranium weight ratio in the ground combination became significantly large beyond 1,000 m.

At the locations where the highest concentrations were detected, chemical dangers were small. As for the radioactive hazard associated with uranium deposited on the ground, a dose of 1,50 mSv

### *Radiological consequences*

The accident and the subsequent evacuation from the controlled zones caused no physical damage to the operating and maintenance personnel.

The systematic air sampling carried out every few hours showed no general contamination of the buildings except for slight air activity in one room of one building. This disappeared as soon as the ventilation system was restarted. The associated surface contamination was cleaned up during the day of 17 April. The air samples also demonstrated that there had been no discharge of radioactivity from the stacks.

### *Restarting conditions*

On 21 April 1980, power was supplied to the plant in the following manner:

- the power supply for the normal operation of the factory was provided from the EDF 90 kV grid by a temporary connection;
- a first emergency power supply was provided by a set of fixed generators capable of maintaining the power necessary for the safety of the installation;
- a second emergency power supply was provided by a set of mobile generators and connection through a temporary line to the EDF 15 kV country grid.

In these conditions, and after regulatory inspection of the system, restart of the plant was authorised on 22 April 1980.

### *The Recuplex Plant, Hanford, 7 April 1962*

This incident occurred when liquid from a sump was collected in a 69-litre, 45.7 cm diameter vessel. The liquid, unidentified at the time, contained between 1,400 and 1,500 grams of plutonium in a volume of about 46 litres after the addition of lean solutions. The only significant exposures were 0.88 and 0.16 Gy received by personnel at distances of about 2.1, 3.2 and 7 m respectively from the excursion.

The site was a plutonium recovery plant in room-sized glove boxes to prevent external contamination. The vessel in which the excursion occurred was normally used for transfer of a dilute side stream for solvent extraction columns to a secondary recovery process, similar to the raffinate-treatment process of the Los Alamos accident. Apparently the concentrated solution had overflowed from a geometrically sub-critical tank and was sucked into the 45.7 cm diameter vessel through a temporary line used for clean-up operations that were still in progress. A total yield of  $8.2 \times 10^{17}$  fissions occurred over 37 hours, with about 20 per cent of the energy released in the first half hour. An initial pulse of approximately  $10^{16}$  fissions was followed by smaller pulses for about 20 minutes, after which boiling occurred, ultimately distilling off enough water to stop the reaction.

The initial pulse, accompanied by the usual blue flash, triggered a radiation alarm, and the area was evacuated promptly, presumably before a second pulse. A unique feature of the analysis of the events was the use of a small, remote-controlled robot developed for handling irradiated fuel. By means of this device, the excursion site was located, meters were positioned and read, and valves were operated.

A new plant to replace Recuplex had been authorised before the accident, and operations were not resumed until it became available. In the modern plant, vessels that are not sub-critical by geometry usually contain neutron absorbers. The system is adaptable to a variety of uses without improvisation, and equipment is easier to keep clean. It is recognised that the flexibility needed in this salvage plant requires special effort to maintain realistic, up-to-date written procedures.

### *Wood River Junction Plant, 24 July 1964*

This accident was initiated when concentrated enriched uranium solution was inadvertently poured into a 45.7 cm diameter (75 litres) tank. The first of two excursions resulted in a lethal exposure and the second, about 2 hours later, was primarily responsible for two other significant radiation doses.

Start-up difficulties in this plant for recovering highly enriched uranium from scrap led to an unusual accumulation of trichloroethane (TCE) solution of low uranium concentration. Small amounts of uranium were recovered by tedious hand agitation of the TCE with sodium-carbonate solution. An easier process was improvised, in which the TCE was treated in the 45.7 cm diameter tank that had been intended only for the make-up of sodium-carbonate solution used in the normal recovery process. Neither the plant superintendent nor one of three shift supervisors was aware of this practice. Meanwhile, solutions of unusually high uranium-235 concentration, resulting from clean-out of plugged equipment had been stored in 11-litre, 12.7 cm diameter bottles identical to those that contained the contaminated TCE. Apparently, a bottle of the concentrated solution was mistaken for TCE and was poured into the sodium-carbonate solution being stirred in the make-up tank. The sock from a single pulse of  $\sim 10^{17}$  fissions knocked the operator onto the floor and splashed part of the solution out of the tank. A flash of light was observed. The victim received an exposure estimated to the 100 Gy and died 49 hours later.

- (18) Report on incidents in Sweden, presented at the annual meeting of CSNI Working Group on the Safety of the fuel Cycle, September 1.
- (19) Health and Safety Executive (1980), *The Leakage of Radioactive Liquor into the Ground at BNFL Windscale*, 15 March 1979.

alarm, presumably were exposed for about 15 seconds, which is roughly the interval between the first two pulses. It is apparent that exposures were limited by prompt evacuation.

The following corrective measures were adopted subsequently. Instead of relying upon valves for isolating equipment, transfer lines that may contain fissile material are actually disconnected. Only vessels that would be sub-critical when containing uranium-235 enriched uranium solutions are permitted.

*The Los Alamos Scientific Laboratory, 30 December 1958*

The next accident resulted from the concentration of plutonium in a solvent layer which was found in a large tank that was supposed to contain only lean aqueous-organic emulsion. A transient change of shape of the solvent layer when a stirrer was started established a criticality of short duration. The result was a fatality and two other significant exposures.

The accident occurred in an area where residual plutonium, usually about 0.1 gram per litre, and americium were recovered from dilute raffinate. Because the normal plutonium inventory was only 0.1 kg, solvent extraction was conducted in large closed tanks. As at Y-12, a material inventory was in progress and it was intended that the tanks be emptied and cleaned individually. Instead, residues and acidic wash solutions from four vessels were combined in a single 850-litre, 96.5 cm diameter tank; many interconnecting transfer lines made this possible. An excursion of  $1.5 \times 10^{17}$  fissions occurred when a stirrer in this tank was started.

As discussed later, a 20.3 cm thick, 160-litre organic layer floating on a dilute-aqueous solution contained 3.27 kg plutonium. It is presumed that the source of this plutonium was solids that had accumulated gradually in the tanks during the 7.5 years of operations and that the organic layer resulted from separation of the emulsion phases by added acids. The initial effect of the stirrer was to thicken the axial part of the organic layer sufficiently for super-criticality. Continued rapid stirring mixed the two phases, diluting the plutonium to a sub-critical concentration.

The operator, who was looking into the tank through a sight glass, received an exposure of 120 ( $\pm$  60) Gy and died 36 hours later. Two men who went to the aid of the victim received doses of 1.3 Gy and 0.35 Gy. There was neither damage to equipment nor contamination, although a shock displaced the tank support 10 mm. A radiation alarm 53 m away was activated and a flash of light was seen from an adjoining room.

The entire recovery plant, which had been scheduled for rebuilding after another six months of operation, was shut down immediately. After ultimate conversion to geometrically sub-critical equipment, the following corrective measures were adopted: written procedures and nuclear safety training were improved, unnecessary solution transfer lines were blocked, and auxiliary vessels such as vent tanks and vacuum buffer tanks were "poisoned" with borosilicate glass raschig rings. Periodic surveys with portable neutron detectors to locate abnormal plutonium deposits were instituted. The accident also led to more complete coverage of process areas by improved gamma-ray sensing radiation alarms.

## 8. Main Conclusions

Since 1981 at the appearance of the previous report on the Safety of the Nuclear Fuel Cycle, the fuel cycle industry has considerably expanded and some sectors have seen their activity multiplied by five. At present the growth rate in the OECD countries seems stabilized and will depend to a great extent on the general evolution of the nuclear power capacity throughout the world. The fuel cycle facilities in operation or under construction can draw upon data from a very extensive and well-documented safety record accumulated over the past 30 to 40 years in international and national publications by technical experts and safety authorities. The major choices with regard to processes have been based on economic and safety considerations.

In the front end of the fuel cycle, no important changes have occurred in the overall production methods leading from uranium ore extraction to the production of fuel elements. However, due to the economic competitiveness, more emphasis was put on the production of  $\text{UO}_2$  and MOX fuel for LWRs at the expense of metallic uranium for the gas-cooled reactors and of  $\text{UO}_2$ - $\text{PuO}_2$  fuel for fast reactors.

From a safety point of view, the attention of OECD Member countries has expanded to include the "in process" problems and worker dose preoccupation as well as the external impact of the facilities on the environment.

The production of mill tailings, which accompanies the extraction of uranium from ore, aroused much attention among the safety authorities and strict regulations have been enacted to reduce the radiological consequences to the environment and to stabilize geochemically the treated ore-residue bodies. The underground mining of rich uranium ores is still a challenge to the personal safety of the workers due to the high radon concentration in the air. The uranium conversion units, from  $\text{U}_3\text{O}_8$  to  $\text{UF}_6$  or vice versa, and the uranium enrichment facilities are well established large-scale industrial processes with very high throughputs. These operations do not constitute a significant radiological hazard, but involve an inherent toxic chemical hazard owing to the massive use of  $\text{UF}_6$  and fluorinated reagents. Such hazards are similar to those in the conventional chemical industry and their control is well-established and understood.

The impact of external hazards (such as earthquakes, fires) on the safety of the large enrichment plants, was recently examined carefully. This examination confirmed the adequacy of the initial safety design concepts.

With the continuing buildup of depleted  $\text{UF}_6$  stocks from some enrichment plants, increased attention is to be given to the safe storage of these residues or to their conversion to less hazardous chemical compounds, i.e.  $\text{UF}_4$ ,  $\text{U}_3\text{O}_8$ , etc. which can be stored without much risk for an extended period of time and which are not hazardous even in accidental conditions such as aircraft crashes, fires, etc.

### Liquid discharges:

The WAK is connected to the Karlsruhe Research Center in respect of the treatment of liquid effluents and the conditioning of waste; therefore, no specific data on liquid discharges are available for the WAK alone.

### Tokai reprocessing plant:

#### Radioactivity released to the air

Nuclide	Annual	Activity, Bq				
		1985	1986	1987	1988	1989
Krypton-85	8.9E + 16	1.0E + 16	1.3E + 16	1.2E + 16	2.7E + 15	9.8E + 15
H <sub>3</sub>	5.6E + 14	2.8E + 12	2.7E + 12	3.7E + 12	2.5E + 12	3.7E + 12
Iodine-129	1.8E + 09	1.0E + 09	2.3E + 08	1.4E + 08	8.5E + 07	2.4E + 08

#### Radioactivity released to the sea

H <sub>3</sub>	1.9E + 15	2.6E + 14	2.4E + 14	2.6E + 14	7.4E + 13	2.4E + 14
Iodine-129	2.7E + 10	8.9E + 07	6.7E + 06	2.0E + 06	nd	1.3E + 07
Total α	4.1E + 08	1.1E + 07	1.9E + 06	1.8E + 07	4.1E + 05	1.1E + 07
Total β	9.6E + 11	3.6E + 07	4.1E + 07	3.7E + 07	8.5E + 06	7.6E + 06
Strontium-90	3.2E + 10	2.3E + 06	2.5E + 07	8.5E + 06	nd	nd
Caesium-137	5.5E + 10	7.8E + 07	1.7E + 08	1.5E + 08	8.5E + 07	4.0E + 07
Plutonium(α)	2.3E + 09	4.8E + 06	9.3E + 06	1.2E + 07	4.1E + 06	4.8E + 06

nd = not detectable  
Total α and β excluding specified single nuclides

## 7.3 Major incidents

As in other areas, incidents and accidents have occurred in the nuclear fuel cycle despite all technical and operational precautions. In all OECD Member countries, these abnormal events are reported by the plant operator to the authorities. Care is taken to collect this information in order to improve safety in the facilities and to provide information to the public and scientific bodies. Since minor incidents are reported and collected in most countries, the public might conclude that the nuclear fuel cycle industry is plagued by a never-ending series of incidents. Certainly it has to be admitted that incidents have occurred and will occur in the future, but compared to other industrial sectors, the safety record of all stages of the commercial nuclear fuel cycle is impressively good.

To allow comparison of incidents and to improve information provided to the public, severity scales for the evaluation of incidents have been established in some countries. However, the following description of major incidents, which have happened in fuel cycle facilities, is not based on a systematic evaluation of the severity of their consequences.

Storage of magnesium clad fuel resulting from the gas-cooled reactors is a declining activity which proved to be acceptable only for a limited period of time due to corrosion of the cladding. Reprocessing of Magnox fuel is therefore a necessity for which industrial plants exist in the United Kingdom and France.

The major hazard associated with Magnox reprocessing is fire; since the cladding is made of magnesium and the core of metallic uranium. During the mechanical decladding operations all possible precautions (active and passive) should be taken to avoid ignition and, in case a fire starts, to mitigate it. In 1987 an extensive regulatory safety audit took place at the Sellafield reprocessing plant and investigated all issues relevant to the safety of the plant and the environment.

In France, a silo filled with magnesium cladding hulls caught fire and therefore all storage facilities are now kept under a water cover. In the United Kingdom the Magnox swarf is being encapsulated in concrete for intermediate storage.

The liquid extraction carried out with TBP dissolved in a kerosene type solvent is the second potential source of fire for which adequate measures should be taken. However until now no large fires have been recorded in the Magnox reprocessing plants.

From a safety point of view, reprocessing of LWR fuel is the most complex part of the entire fuel cycle. Preventive measures against internal potential hazards, e.g. criticality, fire, explosion, corrosion, loss of containment, loss of cooling and mechanical damage, have to be taken into account during the design, construction and operation phases. Recently attention has turned to the external hazards, particularly seismic events and aircraft crashes. However, appropriate site selection and other measures could prevent such hazards to a satisfactory level.

The main potential hazard associated with industrial reprocessing lies in the fact that the nuclear materials are processed in the dispersible form of solution and powder and that industrial quantities of flammable solvent are used. However, only a small fraction of radioactive species is contained in the off-gas and the ventilation, and filtration systems are designed to cope with this. Chemical decomposition of organic reagents, e.g. TBP, degraded solvents, hydrazine, formaldehyde and production of radiolytic gases ( $H_2$ ), are the main explosion risks occurring in the PUREX process. The risks are carefully analysed in order to ensure process control systems capable of detecting and arresting the progress of abnormal events, and further to limit the consequences of an accident to the plant itself. The basic design of the plant must incorporate all measures to avoid criticality in every compartment, for example by geometric constraint and by proper mass flow control.

Fire and explosion risks should be reduced as much as possible and below an allowable value. Its spreading should be restricted by passive measures, e.g. compartmentalisation, use of incombustible structural materials, and mitigated by active measures, e.g. inertisation of equipment, fire-extinguishing system.

Loss of containment is a safety concern which is countered by choosing corrosion resistant materials and by surrounding the nuclear materials by a multi-barrier system consisting of the equipment itself, the containment building and a dynamic barrier with gradual ventilation pressure depression.

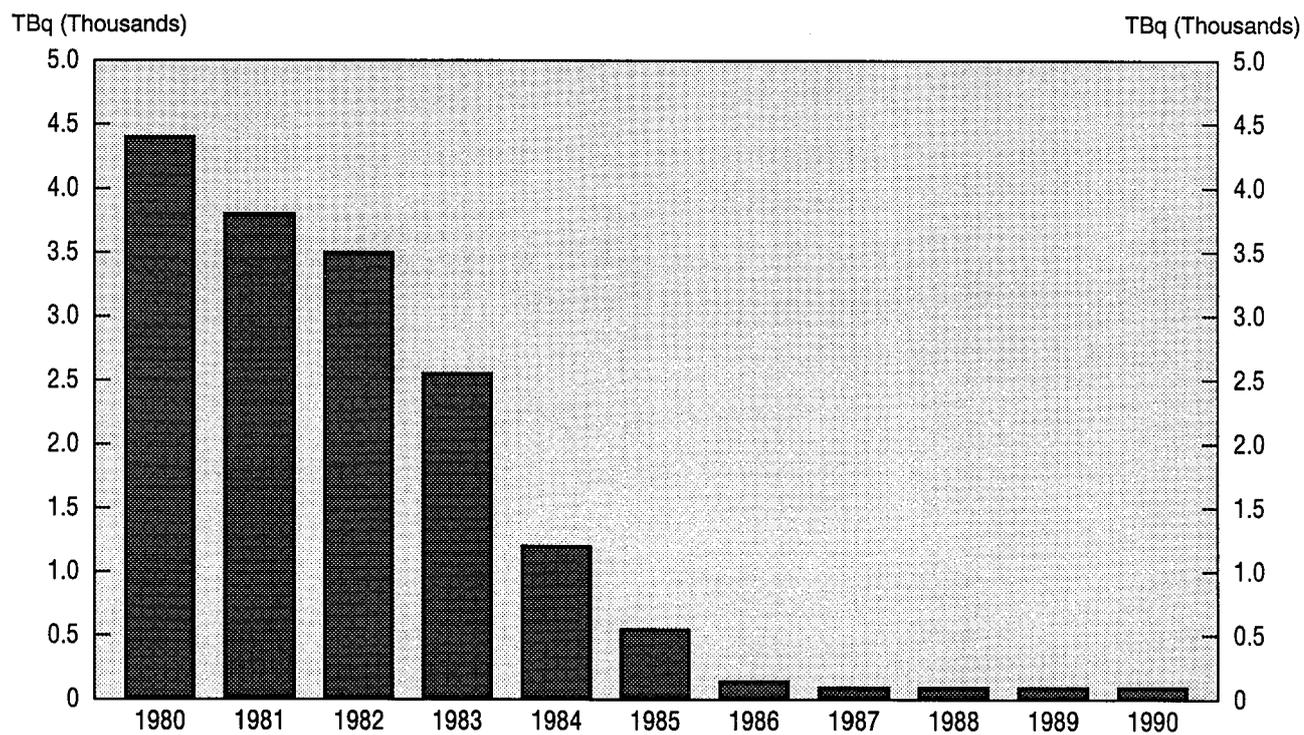
Corrosion resistance of the materials is a very important item in the assessment of the lifetime of the equipment, especially when very corrosive reagents, such as concentrated acids and redox

### Liquid discharges, TBq

Year	H <sub>3</sub>	Total Beta other than H <sub>3</sub>	Strontium-90	Caesium-137	Total Alpha
1984	1,586	1,190	72	434	14
1985	1,062	587	52	325	6
1986	2,150	118	18	18	4.4
1987	1,375	89	15	12	2.2
1988	1,724	81	10	13	2.1

As the following figures demonstrate, a remarkable reduction of total alpha and beta radioactivity discharged by liquid effluents has been achieved in the last decade.

### Liquid discharges from Sellafield (total beta)



principle and the defence in depth concept. The full governmental surveillance on the licensing process of fuel cycle facilities – including siting, design, construction, commissioning, operation and eventual decommissioning – is structured differently in each country, although international and particularly IAEA activities can provide a significant degree of commonality. The follow-up of national and international regulations is achieved by inspectors who check the compliance with the site licence and monitor the licensee's safety performance.

Both internal and external safety aspects have to be taken into account in a comprehensive analysis of the plant safety.

The internal safety aspects to be discussed in a generic way are containment and ventilation, criticality and fire hazards; these are independent of the type of nuclear activity. The emphasis put on each of these aspects depends on the size of the facility and the quantity of nuclear materials involved. The external hazards must take into account the influence of major events (seismic events, floods, etc.) on the integrity of the plant and on the safety of the workers as well as that of the surrounding population.

Technical safety assessments are primarily concerned with the radiological safety. This includes the radioactive risk to man and his environment not only from direct radiation but also from environmental sources. The analysis of the mechanism for abnormal occurrences typical of nuclear facilities, such as criticality accidents, explosions, plant faults is also part of the technical safety assessment.

The man-machine interface is a new important aspect of safety management which ought to be taken into consideration. Though computer control has taken over many tasks in the plant formerly controlled by man, assessment of its reliability and fitness for purpose is essential.

Finally the assessment of accident consequences is also essential to any safety analysis since that aspect is directly connected to the public's perception of incidents and accidents. Recently the IAEA/NEA severity scale was developed for optional use by countries to inform the public about the seriousness of an incident by association with its severity levels which are analogous to the earthquake classification, i.e. the MSK scale.

Some countries have specific severity scales for fuel cycle facilities which have the advantage of being more specific, but which also might confuse the public who tend not to make a clear distinction between nuclear power plants and other nuclear facilities. A uniform severity scale with equivalent steps of impact from each type of nuclear facility is therefore commendable.

Special attention is being given in this report to the safety record of the nuclear fuel cycle facilities in the OECD countries. For reasons of historical overview all the significant incidents in the fuel cycle facilities have been reported according to their date of occurrence from 1950 up to 1990 including those reported in the 1981 report. The very good safety record of the civilian nuclear fuel cycle facilities in the OECD countries is a proof of the adequacy of the current regulatory and operating practices.

The steady decrease in personnel irradiation doses, and in effluent discharges into the surface waters and the air, is an illustration of the continued improvement in operating the facilities and the reduction of the radiological impact of the nuclear fuel cycle on the human environment.

### 7.2.3 Uranium dioxide fuel element fabrication

Generally the gaseous effluents from uranium dioxide fuel element fabrication plants are very small (several grams of uranium per year, radioactivity in the order of 0.01 to 1 GBq alpha activity). The amount of uranium and radioactive daughter products discharged with liquid effluents is largely dependent on whether a wet or dry process is used for the conversion of UF<sub>6</sub> to UO<sub>2</sub>. The liquid discharges are very small for the dry conversion process and for wet conversion in the case of complete conditioning of liquid wastes to a solid CaF<sub>2</sub> waste product. Otherwise the amount of uranium in liquid discharges is dependent on plant throughputs and roughly of the order of some tenths of kilograms of uranium per year (some GBq per year).

### 7.2.4 MOX fuel element fabrication

Due to the high radiotoxicity of plutonium, the off-gas from MOX fuel element fabrication facilities is cleaned by several high-efficiency filters resulting in very small annual releases of radioactivity. As an example, for the new SIEMENS-MOX facility at Hanau, Germany the proposed limits for aerial discharges are 0.55 GBq per year activity and 3.7 TBq per year Rn-220 activity. Also for the MELOX-facility under construction at Marcoule, France, a similar low limit for the discharge of activity is foreseen.

Liquid waste is treated and consolidated to solid waste. For the discharge of remaining liquids or liquid from the uranium section of the MOX facility the proposed limits for discharge at the SIEMENS-MOX facility at Hanau are in the order of tenths of GBq per year.

### 7.2.5 Reprocessing

*La Hague:*

Atmospheric discharges, expressed in Bq (percentage of authorisation):

Year	Gas other than H <sub>2</sub>	H <sub>2</sub>	Halogenes	Aerosols
1982	5.1 × 10 <sup>16</sup> (10.6)	6.3 × 10 <sup>12</sup> (0.29)	1.5 × 10 <sup>10</sup> (14.0)	4.6 × 10 <sup>7</sup> (0.062)
1983	5.0 × 10 <sup>16</sup> (10.4)	8.3 × 10 <sup>12</sup> (0.38)	2.2 × 10 <sup>10</sup> (19.5)	3.9 × 10 <sup>7</sup> (0.053)
1984	2.7 × 10 <sup>16</sup> (5.6)	8.5 × 10 <sup>12</sup> (0.39)	2.8 × 10 <sup>10</sup> (25.2)	3.9 × 10 <sup>7</sup> (0.053)
1985	7.1 × 10 <sup>16</sup> (14.8)	3.3 × 10 <sup>13</sup> (1.50)	2.2 × 10 <sup>10</sup> (19.6)	6.9 × 10 <sup>8</sup> (0.932)
1986	2.9 × 10 <sup>16</sup> (6.0)	6.1 × 10 <sup>12</sup> (0.28)	1.10 × 10 <sup>10</sup> (10.4)	5.5 × 10 <sup>7</sup> (0.074)
1987	3.5 × 10 <sup>16</sup> (7.3)	1.5 × 10 <sup>13</sup> (0.68)	1.5 × 10 <sup>10</sup> (13.2)	5.5 × 10 <sup>7</sup> (0.074)
1988	2.7 × 10 <sup>16</sup> (5.6)	2.1 × 10 <sup>13</sup> (0.95)	2.2 × 10 <sup>10</sup> (19.6)	2.6 × 10 <sup>7</sup> (0.035)
1989	4.2 × 10 <sup>16</sup> (8.8)	2.5 × 10 <sup>13</sup> (1.14)	2.8 × 10 <sup>10</sup> (25.6)	2.3 × 10 <sup>7</sup> (0.031)
1990	6.3 × 10 <sup>16</sup> (13.1)	2.5 × 10 <sup>13</sup> (1.14)	1.9 × 10 <sup>10</sup> (17.4)	2.1 × 10 <sup>7</sup> (0.029)
1991	1.0 × 10 <sup>17</sup> (21.0)	2.8 × 10 <sup>13</sup> (1.27)	2.4 × 10 <sup>10</sup> (21.4)	2.6 × 10 <sup>7</sup> (0.035)

## 9. Fuel Cycle Facility Listing of OECD Countries Status 1992

Sources: *Nuclear Engineering International, Special Edition (1990) World Nuclear Industry Handbook*  
*ESARDA - Analysis of Nuclear Fuel Cycle in E.C. Countries (1988)*  
*IAEA - Nuclear Fuel Cycle Information System (1987)*

### Uranium Ore Processing

Country-Site	Production capacity (THM per year)	Product process
<b>Australia</b>		
Olympic Dam	1,600	Ore - U <sub>3</sub> O <sub>8</sub>
Ranger	2,500	Ore - U <sub>3</sub> O <sub>8</sub>
<b>Belgium</b>		
Puurs-Englis	50	U Ex. H <sub>3</sub> PO <sub>4</sub>
<b>Canada</b>		
Stanleigh	600	Ore - U <sub>3</sub> O <sub>8</sub>
Key Lake	5,700	Ore - U <sub>3</sub> O <sub>8</sub>
Rabbit Lake	5,400	Ore - U <sub>3</sub> O <sub>8</sub>
Cigar Lake	5,400	In preparation
Midwest Joint Venture	1,600	In preparation
Minatco	2,000	In preparation
<b>France</b>		
Bessines	1,500	Ore - U <sub>3</sub> O <sub>8</sub>
Lodeve (St Martin du Bosc)	1,000	Ore - U <sub>3</sub> O <sub>8</sub>
Mailhac	500	Ore - U <sub>3</sub> O <sub>8</sub>
<b>Greece</b>		
Pananesti	150	Ore - U <sub>3</sub> O <sub>8</sub>
<b>Portugal</b>		
Urgeirica	170	Ore - U <sub>3</sub> O <sub>8</sub>

**WAK, Karlsruhe reprocessing plant:**

Fiscal year	1985	1986	1987	1988	1989
<u>Collective doses (man Sv)</u>					
Total	1.11	1.36	1.72	1.30	0.96
Plant personnel	0.83	0.96	0.94	0.76	0.57
Personnel from contractors	0.28	0.40	0.78	0.54	0.39
<u>Average individual dose (mSv)</u>					
Total	2.0	2.53	2.90	2.04	1.53
Plant personnel	2.3	2.63	2.61	2.05	1.61
Personnel from contractors	1.5	2.33	3.18	2.03	1.42
<u>Maximum individual dose (mSv)</u>					
Plant personnel	19.2	27.0	19.6	11.2	8.0
Personnel from contractors	11.5	16.1	20.5	15.8	15.9

**Tokaï reprocessing plant:**

Fiscal year	1985	1986	1987	1988	1989
<u>Collective doses (man Sv)</u>					
Total	0.84	0.73	1.55	4.46 <sup>1</sup>	1.59
PNC staff	0.24	0.28	0.33	0.29	0.24
Personnel from contractors	0.60	0.45	1.22	4.17 <sup>1</sup>	1.35
<u>Average individual dose (mSv)</u>					
Total	0.51	0.49	0.94	1.96	0.70
PNC staff	0.44	0.50	0.61	0.56	0.42
Personnel from contractors	0.54	0.49	1.10	2.36 <sup>1</sup>	0.79

1. Installation of the second pulse filter, replacement of the acid recovery evaporator.

## 7.2 Waste discharges

### 7.2.1 Uranium mining and milling

Annual uranium releases from Canadian uranium processing facilities in kilograms of uranium:

**CAMECO, Blind River UO<sub>3</sub> Refinery**

	1984	1985	1986	1987	1988	1989
Emissions to air	135	75.8	81.2	91.0	86.0	70.0
Emissions to water	33	12.9	6.6	9.0	13.0	11.0

**Enrichment of Uranium (in SWU per year)**

Country-Site	Production capacity (SWU per year)	Product process
<b>France</b>		
Tricastin	10,000,000	Diffusion
<b>Germany</b>		
Gronau	400,000	Centrifuge
<b>Japan</b>		
Ningyo-Toge	200,000	Centrifuge
Rokkasho-Mura	150,000	Centrifuge
	1,350,000 (Project)	Centrifuge
<b>the Netherlands</b>		
Almelo	1,200,000	Centrifuge
<b>United Kingdom</b>		
Capenhurst	950,000	Centrifuge
<b>United States</b>		
Paducah	11,300,000	Diffusion
Porthsmouth	8,000,000	Diffusion

**Fuel Fabrication**

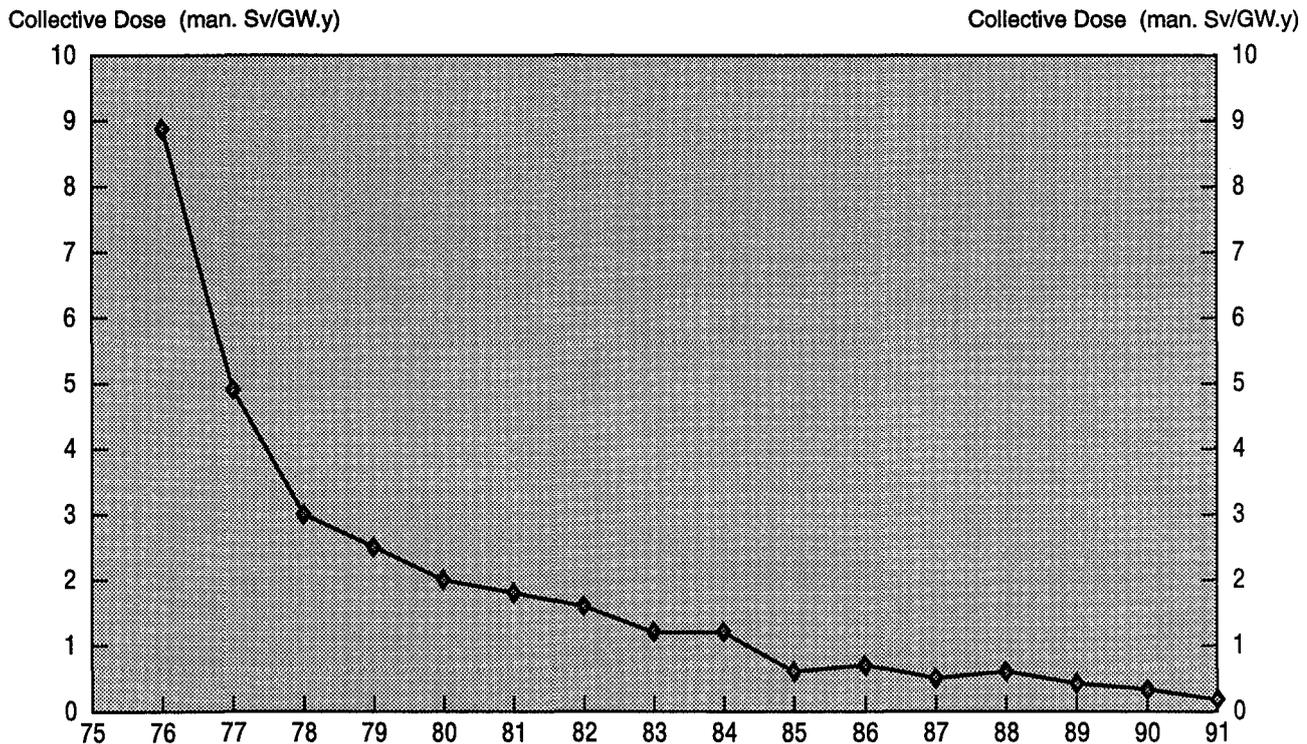
Country-Site	Production capacity (THM per year)	Product process
<b>Belgium</b>		
Dessel	35	MOX-LWR-FBR
Dessel	400	UO <sub>2</sub> - LWR
<b>Canada</b>		
Port Hope	900	Fuel Pellets & Bundles
Peterborough	1,000	Fuel Bundels
Toronto	1,050	Fuel Pellets
<b>France</b>		
Cadarache	25	MOX - FBR
Pierrelatte	500	UO <sub>2</sub> - LWR
Romans	750	UO <sub>2</sub> - PWR
<b>Germany</b>		
Hanau	40	MOX-FBR
Hanau	800	UO <sub>2</sub> - LWR
Karlstein	400	UO <sub>2</sub> - LWR
Lingen	300	UO <sub>2</sub> - LWR

### 7.1.5 Reprocessing

#### *La Hague:*

In 1991, the average individual dose at the reprocessing plants in La Hague was approximately 0.5 mSv per year, whole body dose; the total collective dose of La Hague was approximately 3.4 man Sv per year. These data are valid for all the personnel controlled by the radiation protection staff including personnel from subcontractors. The collective dose covers all activities on the site. A correlation of accumulated collective dose and plant throughput given by the energy equivalent gigawatt-year is shown in the following figure.

**Collective dose of La Hague per gigawatt-year**



#### *Sellafield:*

In the decade 1980-89, the external radiation dose to BNFL radiation workers at Sellafield has steadily decreased. In 1989 the average individual dose from external radiation was 3 mSv per year, the collective dose for all Sellafield's radiation workers summed up to 20 man Sv.

## Away from Reactor Storage of Spent Fuel

Country-Site	Storage capacity (THM)	Storage means
<b>France</b>		
Cadarache (Cascad)	180	Vault
La Hague	8,400	Pool
<b>Germany</b>		
Gorleben	1,500	Dry Cask
Karlsruhe	55	Pool
Ahaus	1,500	Dry Cask
Kreifswald	500	Dry Cask
<b>Japan</b>		
Tokai-Mura	97	Pool
<b>Sweden</b>		
(Clab)	3,000	Pool
<b>United Kingdom</b>		
Sellafield (Magnox)	3,500	Pool
Sellafield (LWR + THORP)	5,300	Pool
Sellafield (AGR)	2,200	Pool
Wylfa	700	Vault
<b>United States</b>		
Morris	750	Pool
(MRS)	15,000 (Planned)	Dry

## Reprocessing plants

Plant	Beginning of construction	Starting operation	Capacity THM/YR
La Hague (UP3)	1982	1990	800
La Hague (UP2 800)	1982	1993	800
Sellafield (THORP)	1985	1992	1200
Rokkasho Mura	1992 (planned)	1999 (planned)	800

\* Construction of WAW plant was stopped in 1989.

### 7.1.3 Uranium dioxide fuel element fabrication

#### Springfields fuel manufacturing plant, the United Kingdom

Year	Average individual dose mSv
1986	3.30
1987	3.05
1988	2.45
1989	2.00

#### SIEMENS Uranium fuel element fabrication, Hanau, Germany

##### Radiation exposure in 1989:

	Average individual dose, mSv	Collective dose man mSv
External gamma dose	1.17	568
Effective dose from incorporation	1.07	618
Additional incorporation from abnormal operation		348

### 7.1.4 Mixed oxide fuel element fabrication

#### ALKEM-SIEMENS-MOX fuel fabrication, Hanau, Germany

Year	Average individual dose, mSv	Collective dose man Sv
1984	3.01	1.36
1985	2.15	1.16
1986	4.03	2.63
1987	3.81	2.66
1988	3.41	1.76

**Fuel Fabrication (THM per year)**

Belgium	435	Japan	1,655 (LWR) + (MOX)
Canada	2,900	Spain	200
France	1,775	Sweden	400
Germany	1,540	United Kingdom	2,106
Italy	260	United States	3,625

**Spent Fuel (AFR/THM)**

France	11,380	Sweden	3,000
Germany	3,095	United Kingdom	11,700
Japan	97 (under construction) 3,000	United States	750 (15,000)

**Reprocessing (THM per year)**

France	2,258	United Kingdom	2,700 (Magnox, THORP, etc.)
Japan	1,010 (operation) 210 (under construction) 800		

## **7. Safety Records of Fuel Cycle Facilities**

The experience gained from plant operations constitutes a valuable basis for the analysis and further improvements of safety. The emphasis of this status report is directed to the nuclear and radiological safety of the different types of fuel cycle facilities. It is not within the scope of the report to present a complete compilation of the vast experience and all the safety records of the nuclear fuel cycle facilities. However, operating data should demonstrate what has been achieved in the protection of workers against radiation and in limiting and controlling radioactive effluent discharges from the facilities.

As in the 1981 status report, a review of relevant incidents which have occurred in fuel cycle plants is presented.

It has to be emphasized that the material of this section should be regarded as additional information to the reader of this report to provide some perspective. However the material presented certainly is not sufficient, without considering additional factors, to derive conclusions of the safety status of specific facilities or to make comparisons. Important additional factors are the type and throughput of radioactive material in the facilities, the specific design and operation of the plant and site characteristics.

## Glossary of Terms

### ***Actinide series***

The group of elements from actinium (atomic number 89) to lawrencium (atomic number 103), which together occupy one position in the periodic table; the series includes the naturally occurring and man-made isotopes of thorium and uranium and of the higher actinides such as neptunium, plutonium and curium.

### ***Activation products***

Nuclides which have become radioactive in consequence of bombardment by neutrons or other nuclear particles.

### ***Alpha particle***

A positively charged particle emitted in the decay of some radioactive nuclei, especially nuclides of the actinide series; it consists of two protons and two neutrons and is identical with the nucleus of the helium-4 atom. It has a very low penetrating power and hence pure alpha emitters are almost exclusively only a hazard when taken into the body.

### ***Becquerel***

A unit of radioactivity, defined as 1 disintegration per second. Related to the old unit of Curie (Ci) by  $1\text{Ci} = 3.7 \times 10^{10}$  disintegrations/second;  $1\text{Bq} = 2.703 \times 10^{11}$  Ci.

### ***Beta particle***

An electron or positron emitted in the decay of some radioactive nuclei; It is only moderately penetrating.

### ***Biological shield***

Thick walls, usually of concrete, separating a source of radiation from personnel and designed to reduce the level of the radiation to an acceptable level; the term is applied particularly to the shield surrounding the core of a nuclear reactor with the object of absorbing neutrons and gamma radiation.

### ***Breeder reactor***

A reactor that creates more fissionable fuel than it consumes; The new fissionable material is created by capture in fertile materials of neutrons from fission.

- (17) *Research and Development of Spent Fuel Casks and the Criteria for Seagoing Vessels carrying them*, International Conference on Nuclear Power and its Fuel Cycle, Salzburg, 1977.
- (18) IAEA (International Atomic Energy Agency) Technical Reports Series N°. 229 (1983), *Handling and Storage of Conditioned High Level Wastes*, Vienna.
- (19) VENET, P., et HAYTINCK, B., (1988), *Coûts et modes de financement de l'évacuation géologique des déchets radioactifs*, EUR 11837 FR.
- (20) WARD, T., et al, (1990), *The WIPP Transportation System, Dedicated to Safety, Waste Management '90*, Vol. 1, pp. 235-238.
- (21) WEINER, R.F., et al., (1990), *Risks in Transporting Radioactive Materials, Comparison of Data with NUREG-0170 Projections*, Waste Management '90, Vol. 1, pp. 221-226.
- (22) GOODMAN, L.S., (1990), *Routing Guidelines for Rail Transport of Radioactive Materials, Is consensus possible?*, Waste Management '90, Vol. 1, pp. 227-229, edited by Post & Wacks.

### ***Disposal***

The release or emplacement of waste materials without the intention of retrieval.

### ***Dose***

The mean energy imparted by radiation per unit mass of matter; it is expressed numerically in grays, symbol Gy, as the unit of radiation equal to the joule per kilogram.  $1 \text{ Gy} = 1 \text{ J/kg} = 100 \text{ rad}$ .

### ***Dose equivalent***

The amount of absorbed radiation per unit mass of matter; it is expressed numerically in sieverts, symbol Sv, as the unit of dose equivalent.  $1 \text{ Sv} = 100 \text{ rem}$ .

### ***Dose equivalent limits***

The radiation exposure limits recommended by the International Commission on Radiological Protection (ICRP).

### ***Fast reactor***

A reactor in which the majority of fissions are produced by neutrons having energies of the same order of magnitude as those which they possess when they are produced in the fission process; often quoted as FBR for fast breeder reactors.

### ***Fissile material***

Material capable of undergoing fission.

### ***Fission***

The splitting of a nucleus usually into two approximately equal fragments; the process is accompanied by the emission of neutrons and the release of energy. Neutron-induced fission is the most important, but fission of certain nuclides may occur spontaneously.

### ***Fission products***

Nuclides produced in fission, either directly or by the disintegration of the fission fragments.

### ***Fission reactor***

A reactor in which energy is generated by the fission of heavy nuclei; includes both thermal and fast reactors.

### ***Fuel cycle***

The various stages involved in supplying fuel for nuclear power reactors and any subsequent treatment and disposal operations; it includes uranium mining, milling and enriching, fabrication of fuel elements, their use in a reactor, fuel reprocessing, waste storage and waste disposal.

### ***Gamma-rays***

Electromagnetic radiation with short wavelength ( $10^{-8}$  to  $10^{-11}$  cm) which is emitted by the nucleus. The emission of gamma-rays accompanies the disintegration of many alpha or beta emitters. The penetrating power of gamma-rays is a function of their energy.

hours later and rested on a 15 m deep sand bank. It was loaded with 30 packages of the 48 Y type filled with natural UF<sub>6</sub> (236 THM), and 22 packages type 30B which were empty.

The authorities decided to recover the packages. Due to bad weather the operation took more than one month (till 4 October 1984). The packages were slightly dented by the stormy sea but remained tight except for one package which was damaged at the valve. The protective covers surrounding the valves did not withstand the incident. The contents of the damaged package was checked and showed a loss of 50 kg U.

During the salvage operation, U and F analyses were carried out continuously but no abnormal concentrations were detected. As a result of this accident an important regulatory action was undertaken at the international level and particular attention was drawn to the chemical hazard of UF<sub>6</sub>. A new regulation for UF<sub>6</sub> transport is being reviewed by IAEA (9).

The main new provision prescribes an experimental test on the fire resistance of UF<sub>6</sub> packages (30 minutes at 800°C) and on its behaviour during an impact (crash from 1.8 m height). The latter provision will impose changes to the valve covers, which did not survive the accident. A parallel action has been undertaken by the European Community in order to regulate the stowing and the fastening of large packages inside the shiphold.

**Radioactive waste**

Any material containing or contaminated with radionuclides at concentrations greater than the values that the competent authorities would consider acceptable in materials suitable for uncontrolled use or release, and for which there is no foreseen use.

**Radioactivity**

Process whereby certain nuclides undergo spontaneous disintegration in which energy is liberated, generally resulting in the formation of new nuclides. The process is accompanied by the emission of one or more types of radiation, such as alpha particles, beta particles and photons (electromagnetic radiation).

**Radiolysis**

The decomposition of material by ionising radiation.

**Reprocessing**

A chemical process, the purpose of which is to extract for further use uranium and plutonium from spent fuel; this operation also results in the separation of radioactive waste products.

**Separative work**

A term used in the process of isotope enrichment; it is in relation to the amount of work necessary to increase the isotopic content of natural uranium in uranium-235.

**Sievert (Sv)**

A dose equivalent unit used in radiological protection. Doses in Sv are obtained by multiplying doses expressed in Grays by appropriate factors that take into account the biological effects of the various types of radiation, the dose distribution within the body and any other necessary corrections. The SI systems and ICRP proposed the use of the sievert as a new dose equivalent unit replacing the rem (R);  $1 \text{ rem} = 0.01 \text{ Sv} = 10 \text{ mSv}$ .

**Specific activity**

The activity per unit mass or volume.

**Spent fuel**

Same as irradiated fuel.

**Storage**

The emplacement of waste materials with the intent and in such a manner that the material can be retrieved later.

**Thermal reactor**

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

### *Heat transfer*

In dry transport, heat is mainly transferred by radiation (~ 10% only by convection). The cladding temperature essentially depends upon surface state and emissivity level. 400°C is typical for current designs. In wet transport the cladding temperatures are usually kept below 150°C by either natural or forced convection cooling; the former is, however, preferred in the majority of cases. Heat is transferred to the flask surroundings by natural convection and radiation.

Sometimes materials which are included in flask designs for radiation shielding also serve for fire protection purposes. Shielding consisting of wet plaster or wet, aluminous cement layers is of this type. If the flask is involved in a fire, the wet material is dehydrated and an efficient insulating layer is formed.

### *Radiation shielding*

Lead is often used in spite of its unfavourably low-melting point (327°C). Cast iron or steel are also used, but their use can lead to high weight and large dimensions. The flask opening is usually closed by a barrel or sliding door. Further shielding is provided by the water (in wet flasks), resin or polyethylene.

### *Mechanical design*

The main steel-flask envelope provides the necessary resistance to impact and puncture. Shock-absorber devices are placed at each end of the flask to reduce the consequences of impact. Wash-out plugs are protected by covers.

Transport of spent fuel elements in canisters by truck or by railway wagon is presently the most widespread industrial practice within the continental zones (the United States, Europe) (14). International and national regulations are worked out in detail to prevent radiation hazard in case of incident or accident (1).

Increased attention has recently been given to transport by ship or barge in the United Kingdom, Canada, Sweden etc., as a safer and less-costly alternative to road and rail transport. Ships owned and operated by nuclear fuel cycle organisations or nuclear power plants provide increased safety since they are specially built or modified for these types of transport. Trained personnel present on the ships is an additional safety asset (14, 15, 16).

Safety studies on abnormal events occurring during ocean transport have been carried out in Japan (17).

## **6.4 Transport of vitrified high level waste**

The problems encountered with the transport of vitrified HLW are very similar to those of spent fuel elements: high gamma radiation and heat output. Though the total radioactivity of vitrified HLW canisters is very high (up to 17 000 TBq/canister) the mean gamma energy is lower than that of an equivalent amount of spent fuel and has an energy peaking around 0.6 MeV. The neutron flux resulting from the alpha activity in the glass is taken into account when designing the canister shielding.

Modified spent fuel flasks, e.g. CASTOR 2A, TN 1300 and TN 12.2, among others, are well-suited for transportation of vitrified HLW (18, 19).

## List of abbreviations

<b>AFR</b>	Away From Reactor
<b>ALARA</b>	As Low As Reasonably Achievable
<b>AR</b>	At Reactor
<b>BWR</b>	Boiling Water Reactor
<b>CANDU</b>	Canada Deuterium Uranium
<b>DBE</b>	Design Basis Earthquake
<b>DIN</b>	Deutsche Industrie Norm
<b>DWL</b>	Demand Working Level
<b>FBR</b>	Fast Breeder Reactor
<b>FP</b>	Fission Products
<b>GCR</b>	Gas Cooled Reactor
<b>HAZOPS</b>	Hazard and Operability Studies
<b>HLLW</b>	High Level Liquid Waste
<b>HTGR</b>	High Temperature Gas Reactor
<b>HAZOPS</b>	Hazard and Operability Studies
<b>IAEA</b>	International Atomic Energy Agency
<b>ICRP</b>	International Commission on Radiological Protection
<b>IDR</b>	Integrated Dry Route
<b>KfK</b>	Kernforschungsanlage Karlsruhe
<b>LWR</b>	Light Water Reactor
<b>MOX</b>	Mixed Oxide

Test specifications for type B packages are related to potential accident situations and the package is required to be so designed that, if it were subjected to these tests, the integrity of the containment of the radioactive contents and eventually of the shielding would be retained to the extent specified in the IAEA regulations, in which the nature of the contents is taken into account.

The safety tests on type A packages are based on the premise that the package may be destroyed in a severe accident but require that it withstands normal transport conditions and foreseeable accidents such as a drop in handling, whereas the regulations define the type B package as being able to withstand severe tests corresponding to accident conditions in transport with reasonable safety margins. Free drop impact and fire are of prime concern for type B package tests:

- The drop test corresponds at least to the stresses sustained by the package in the collision of the shipping vehicle on a bridge pile or a concrete wall at a speed of 100 km per hour. The validity of this test has been endorsed by several accidents, such as a drop of enriched UF<sub>6</sub> containers from a carriage onto the track at a speed of 100 km (Germany, 1972 and the United States, 1977), the drop of a spent fuel flask from a highway bridge (the United States, 12 December 1970), and from a crane (Belgium, 1973). Sandia Laboratories at Albuquerque, New Mexico (the United States) carried out in 1977, full-size accident tests involving spent fuel flasks, and truck and locomotive collisions at speeds higher than 100 km per hour (5). In the United Kingdom in 1984, a full-size test involved a train at speed of 160 km per hour crashing into a spent fuel element canister. The test showed that the canister could withstand such an impact (6).
- The fire or thermal test has been specified on the basis of the statistics of airplane and oil tank lorry fires. These conditions are unusual in practice and very difficult to achieve experimentally. Present insulation materials allow packages to withstand fires more severe than those specified in the test regulations.

The transport of fissile material requires consideration of any problems which might arise in the event of a criticality accident. The IAEA regulations require that fissile material packages have criticality approval and this approval must take into account the results of both the various tests to which the packages have been subjected and the accident assessments. A mechanical impact which could bring into close proximity several nuclearly safe quantities of fissile materials by deforming the packages typifies the remote accident situations which have to be considered for type B packages in the criticality safety clearance context.

Practical accident scenarios severe enough to damage the containment and give rise to contamination of the environment are difficult to conceive for adequately designed and maintained packages. Contamination levels would depend on the nature and radioactive inventory of the contents and the possibilities for operator intervention (7, 8). (No environmental contamination resulted from any of the fuel cycle transport accidents listed in Table 6.3)

The possibility of a ship carrying radioactive materials accidentally set on fire or sinking after a collision or another failure has also been considered. Risks from fire and explosion on board of the vessel can be mitigated by adequate protective measures. Even if a ship were to sink, most packages would survive unless the depth exceeds 200 m, however, for the most hazardous materials, packages would survive in much deeper water. The salvage of the packages in the Mont Louis incident shows that, in relatively shallow waters, packages can be recovered virtually undamaged (see 6.5).

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## 6. Transport of Radioactive Materials

The OECD Member countries with a nuclear programme have more than 30 years of experience in the transport of radioactive materials and a great deal of statistical data already exists on the subject.

Specific safety regulations are well-developed and harmonised internationally since they are all based on the IAEA recommendations (1, 2, 3). Accident situations have been taken into account from the beginning and the transport regulations are framed so that unacceptable radiation exposures should not occur from radioactivity release even in the case of severe accidents. Test procedures verify that actual designs comply with these regulations and Table 6.1 cites these for the main types of packages.

Table 6.1 Main tests for packages

Type A package	<ol style="list-style-type: none"><li>1. Free drop test = 1.2 m</li><li>2. Water spray test</li><li>3. Compression test</li><li>4. Penetration test</li></ol>
Type B package	<ol style="list-style-type: none"><li>1. Drop of 9 m onto an unyielding target</li><li>2. Drop of 1 m onto a bar end</li><li>3. Thermal test at 800°C for 30 minutes</li><li>4. Water immersion test under 15 m</li></ol>

The experiences acquired over the years were reviewed thoroughly during the Tenth International Symposium on the Packing and Transport of Radioactive Materials which took place in Yokohama, Japan in September 1992. As far as is known, no accidents have ever been encountered with serious radiological consequences.

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# **THE SAFETY OF THE NUCLEAR FUEL CYCLE**

**The nuclear fuel cycle covers the procurement and preparation of fuel for nuclear power reactors, its recovery and recycling after use and the safe storage of all wastes generated through these operations.**

**The facilities associated with these activities have an extensive and well documented safety record accumulated over the past 40 years by technical experts and safety authorities. This report constitutes an up-to-date analysis of the safety of the nuclear fuel cycle, based on the available experience in OECD countries. It addresses the technical aspects of fuel cycle operations, provides information on operating practices and looks ahead to future activities.**